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Title: MCNP6 Delayed Neutron Emission Validation with Experimental Measurements

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Intended for:

MCNP6 Delayed Neutron Emission Validation with Experimental Measurements

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Dr. Emily Corcoran, RMC
Dr. David Kelly, RMC
LA-UR 11-05868

October 11 2011

Overview

- Introduction, Background & Theory
- Experimentation
- MCNP6 Model
- Results & Discussion
- Conclusions

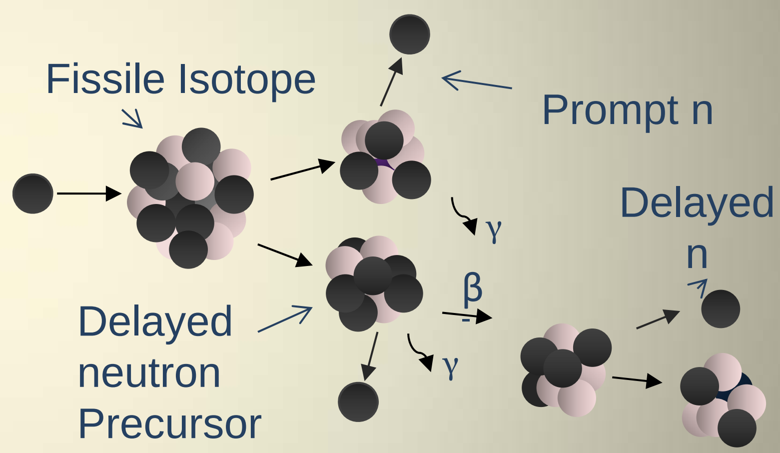
The Delayed Neutron Counting System

- Constructed in 2010
- Validated for ^{235}U analysis, aqueous samples
- Current experiments include ^{233}U samples
- Expanding analysis to include ^{239}Pu analysis and mixtures of two or more fissile isotopes
- Winter 2011/2012 will be updated with new hardware to increase sensitivity and efficiency

Delayed Neutron Generation

- Many delayed neutron precursors

Group	$t_{1/2}$ [s]	λ [s^{-1}]	$\beta_i = \nu_i/\nu_{d\,[\%]}$
1	55.6	0.014267	0.0328 ± 0.0042
2	24.5	0.028292	0.1539 ± 0.0068
3	16.3	0.042524	0.091 ± 0.009
4	5.21	0.133042	0.197 ± 0.023
5	2.37	0.292467	0.3308 ± 0.0066
6	1.04	0.666488	0.0906 ± 0.0046
7	0.424	1.634781	0.0812 ± 0.0016
8	0.198	3.554600	0.0229 ± 0.0095



Delayed Neutron Generation

- Eight (or 6) groups, denoted by $t_{1/2}$ and production ratio

Group	$t_{1/2}$ [s]	λ [s^{-1}]	$\beta_i = \nu_i / \nu_{d}$ [%]
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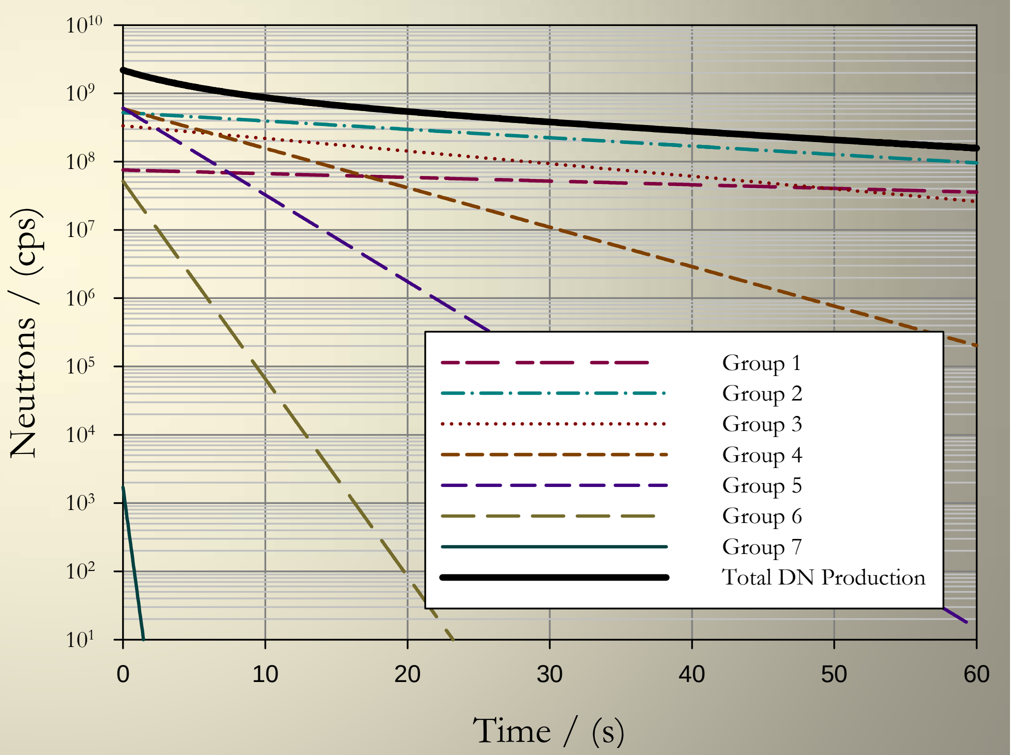

Delayed Neutron Generation

- Production ratios dictated by fission fragment yields

Group	$t_{1/2}$ [s]	λ [s^{-1}]	$\beta_i = \nu_i/\nu_{d}$ [%]
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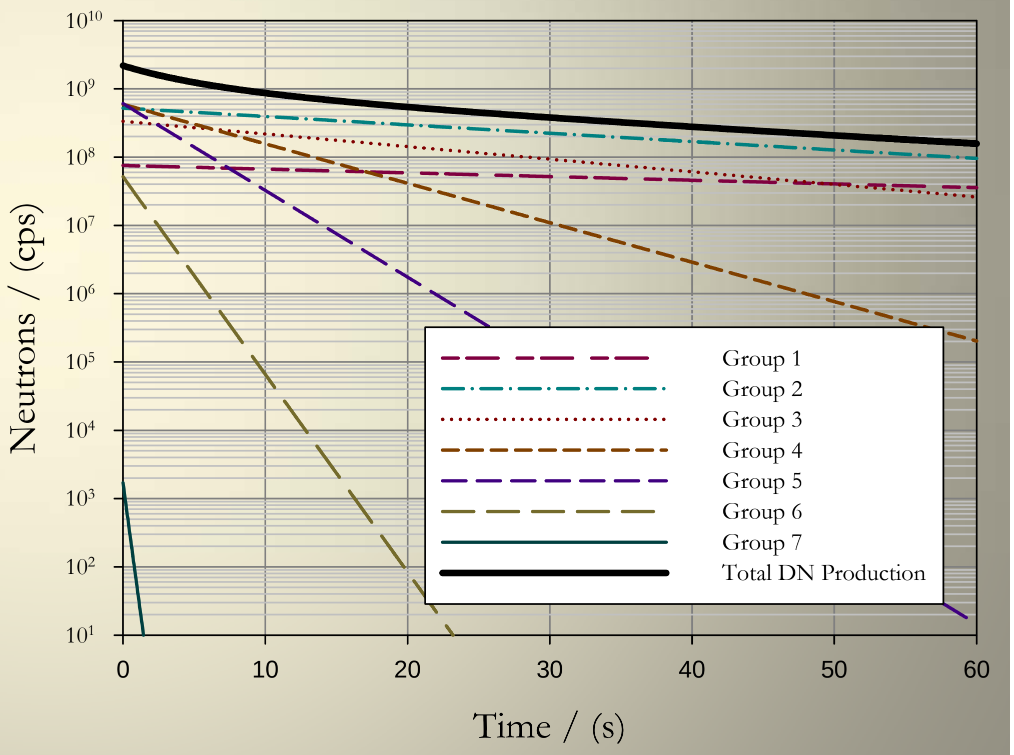
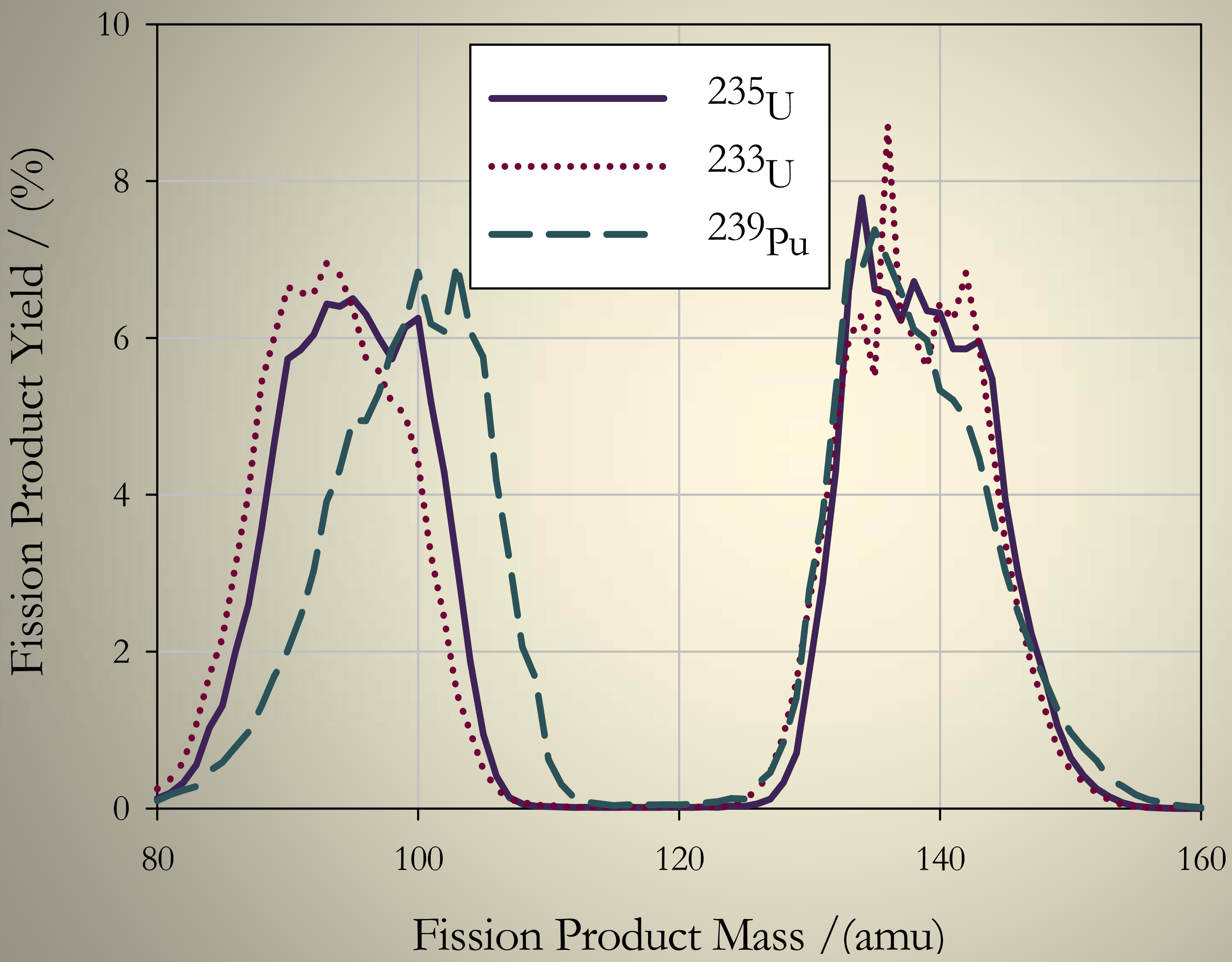
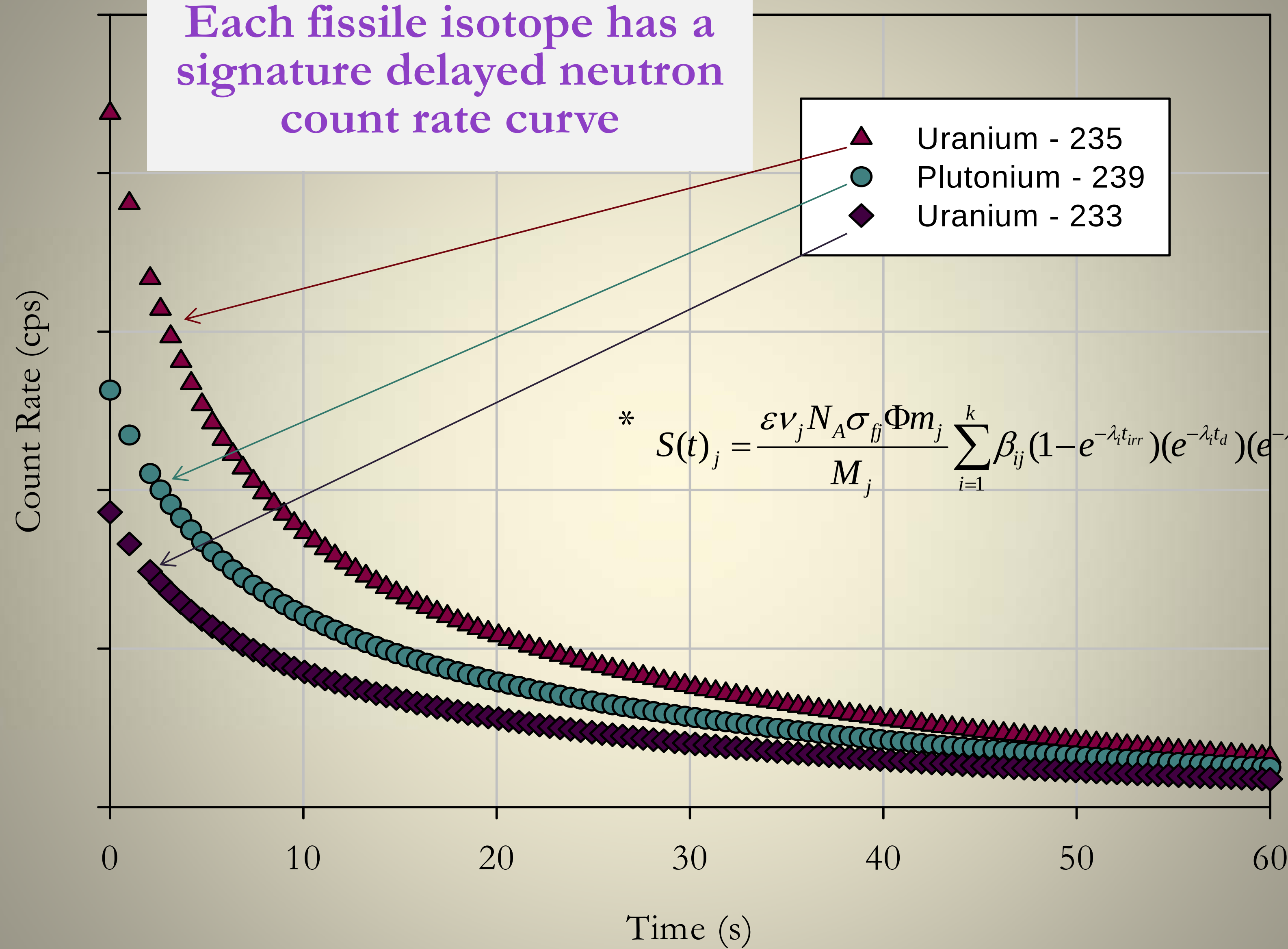




$\beta_i = \nu_i/\nu_{d\,[\%]}$
0.0328 + 0.0042
0.1539 ± 0.0068
0.091 ± 0.009
0.197 ± 0.023
0.3308 ± 0.0066
0.0906 ± 0.0046
0.0812 ± 0.0016
0.0229 ± 0.0095

Each fissile isotope has a signature delayed neutron count rate curve



$$*\quad S(t)_j = \frac{\varepsilon \nu_j N_A \sigma_{fj} \Phi m_j}{M_j} \sum_{i=1}^{k} \beta_{ij} (1 - e^{-\lambda_i t_{irr}})(e^{-\lambda_i t_d})(e^{-\lambda_i t})$$

*Li X, Henkelmann R, Baumgartner F (2004) Nucl Instrum Methods Phys Res B 215:246–251

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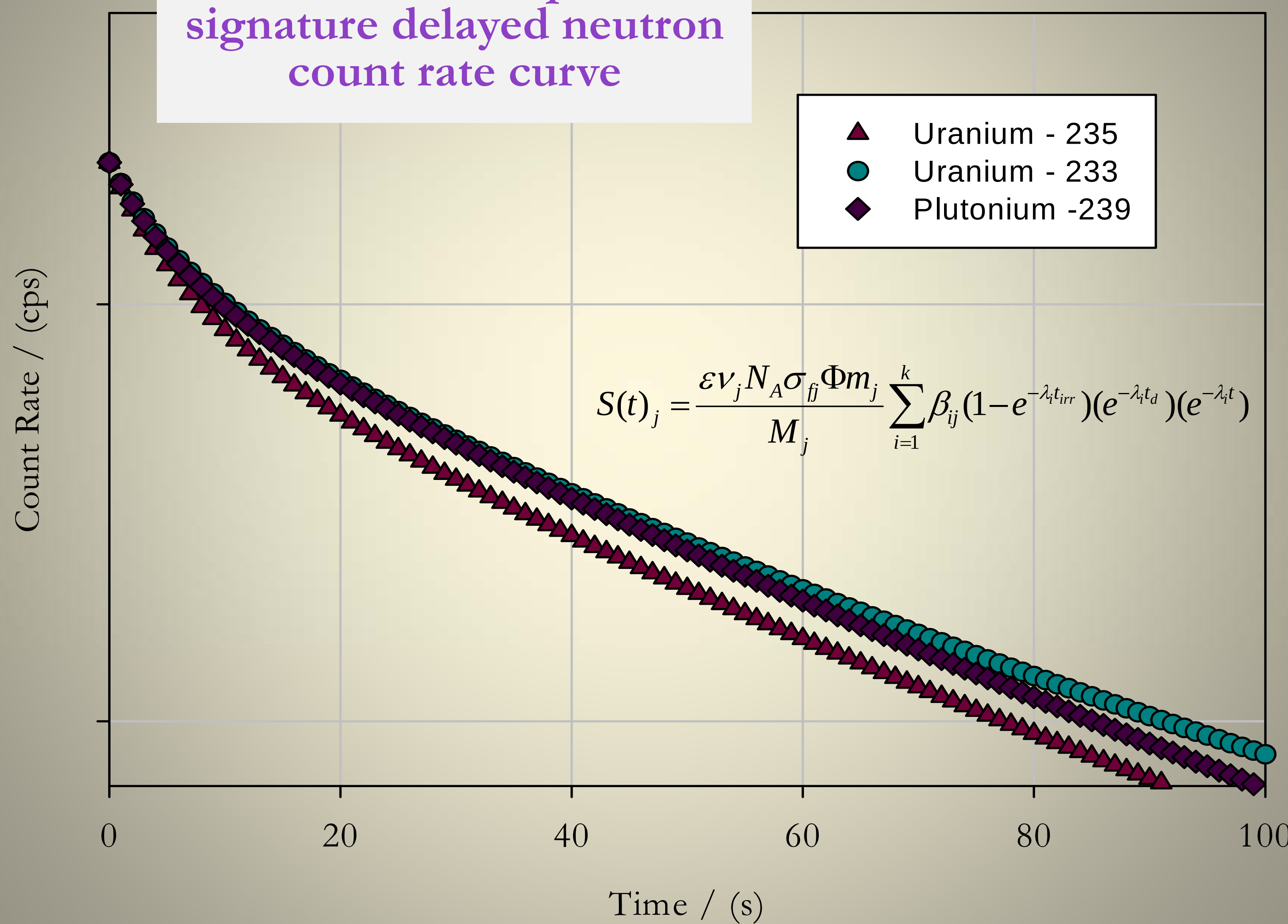

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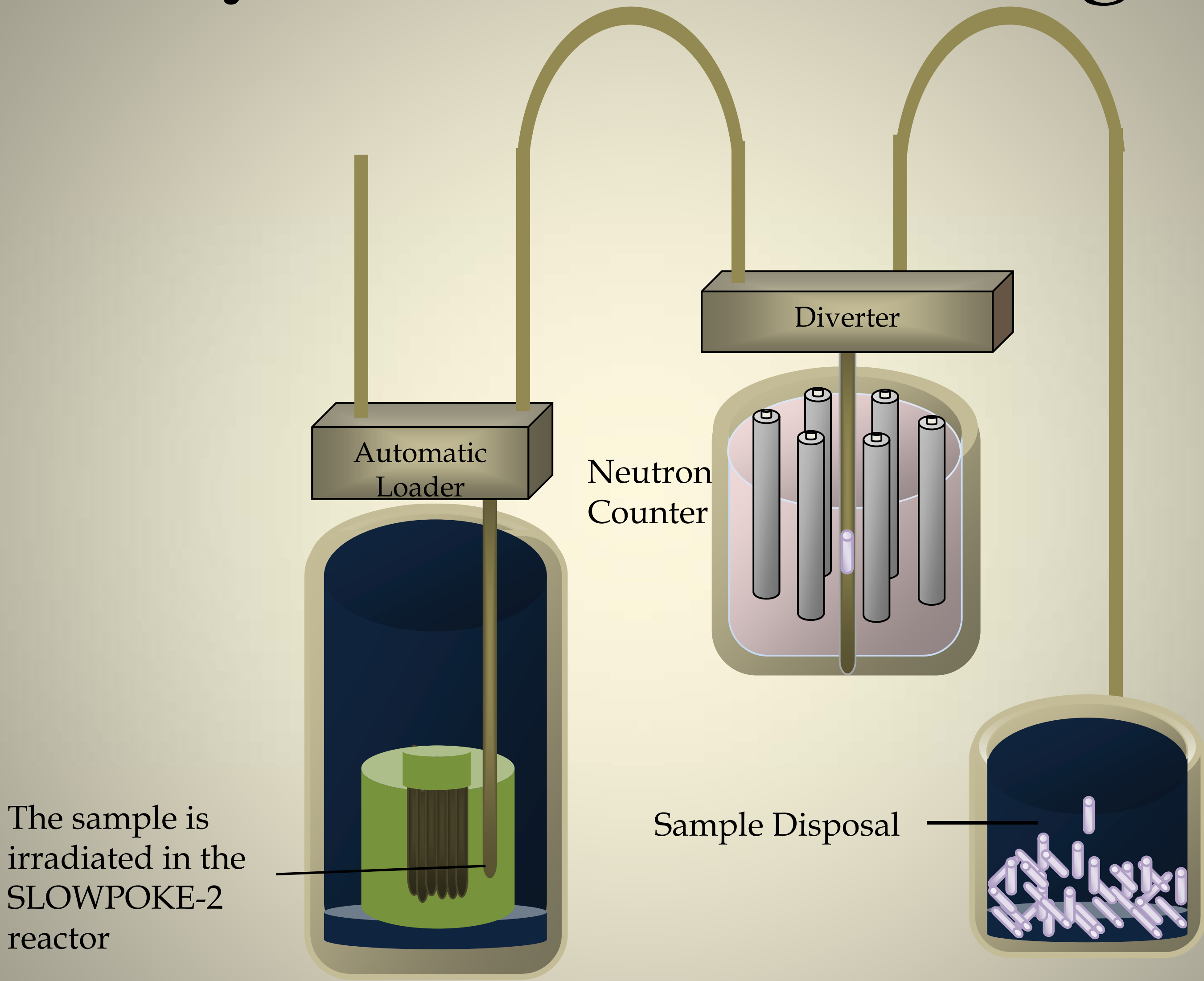
The Delayed Neutron Counting System

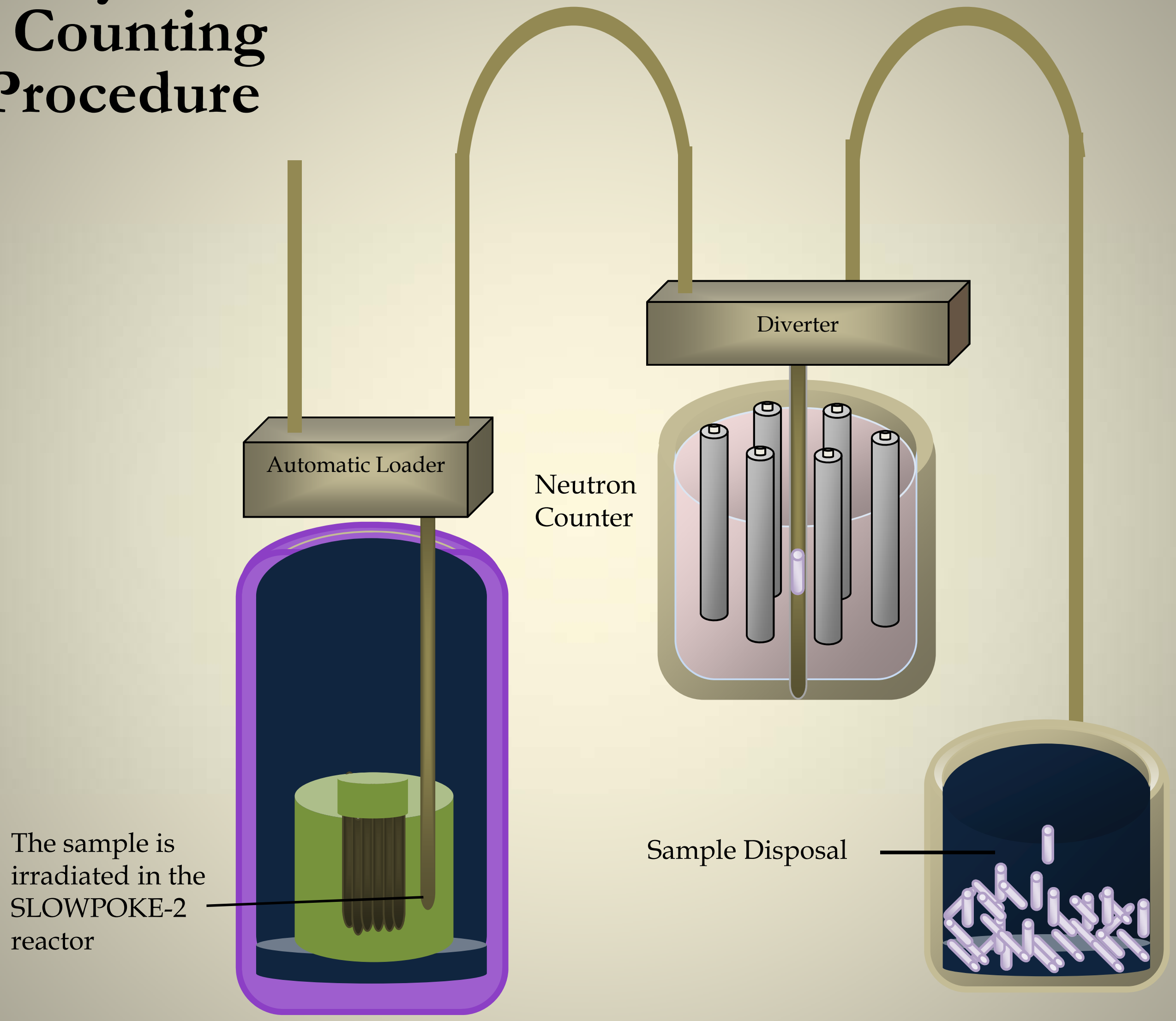


Sample Preparation

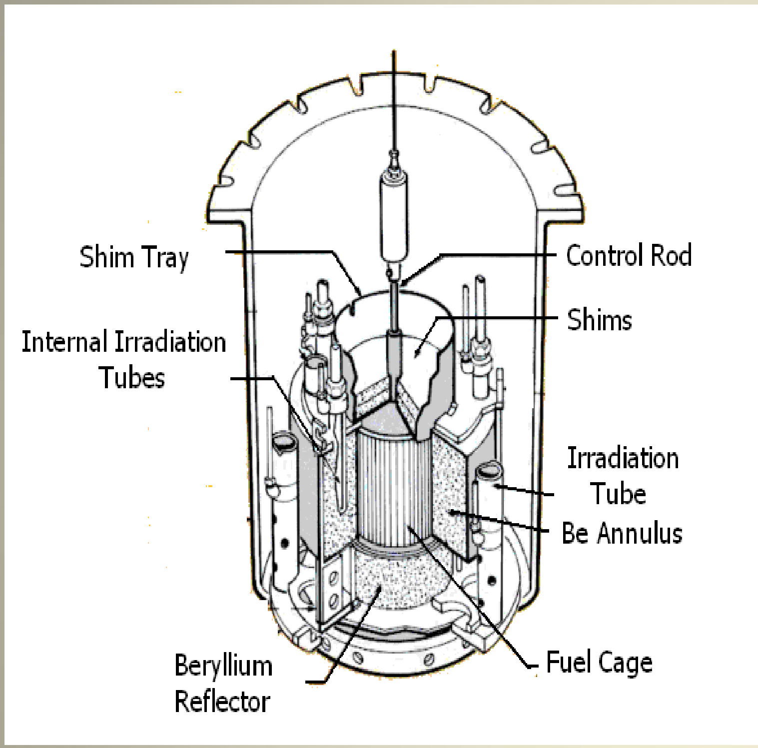
2.25 "

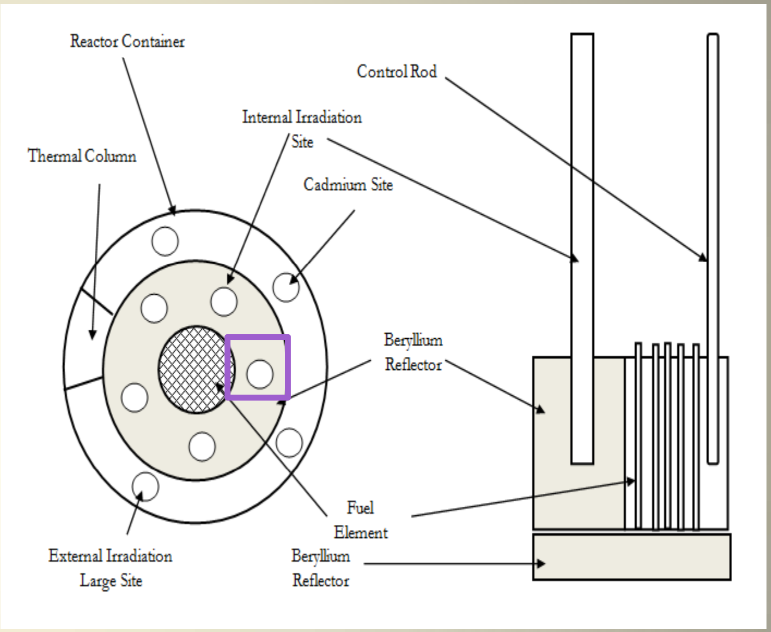
0.67"

The Delayed Neutron Counting System Procedure



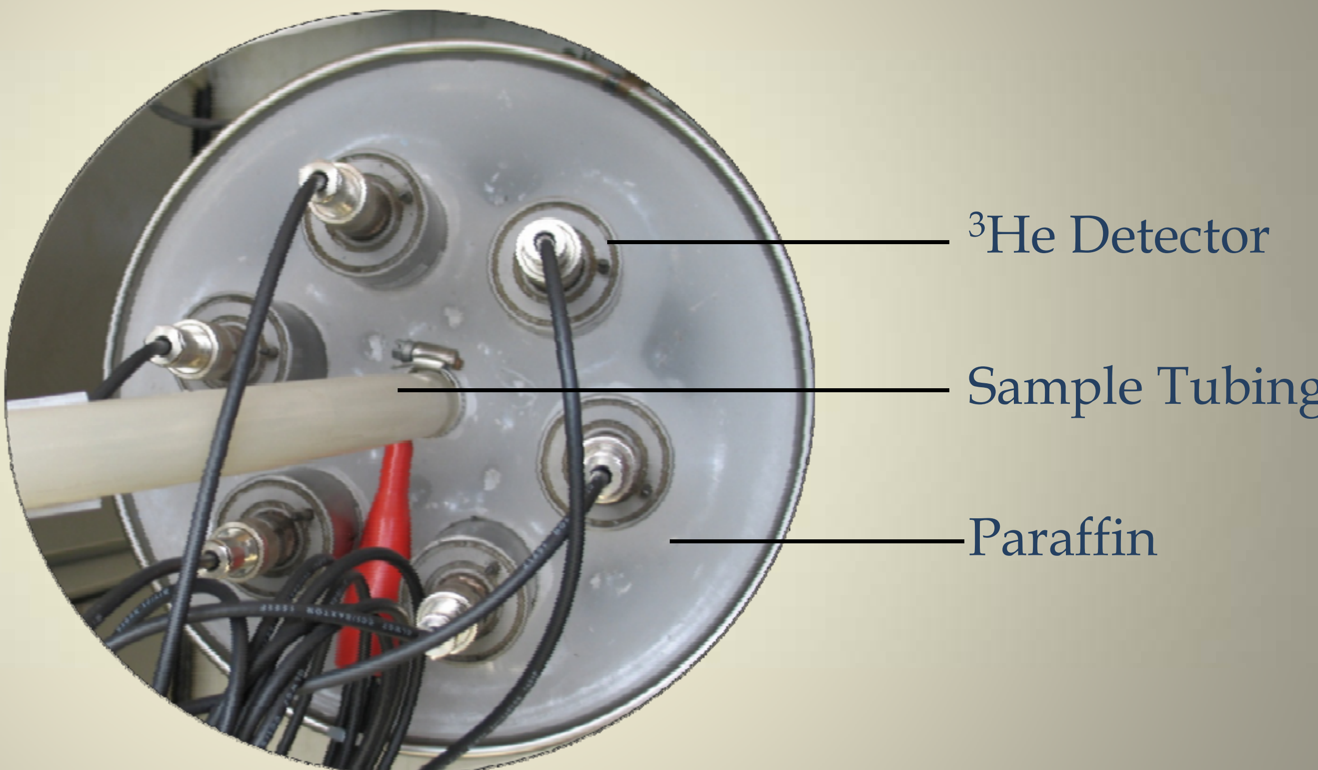
The SLOWPOKE-2 Reactor



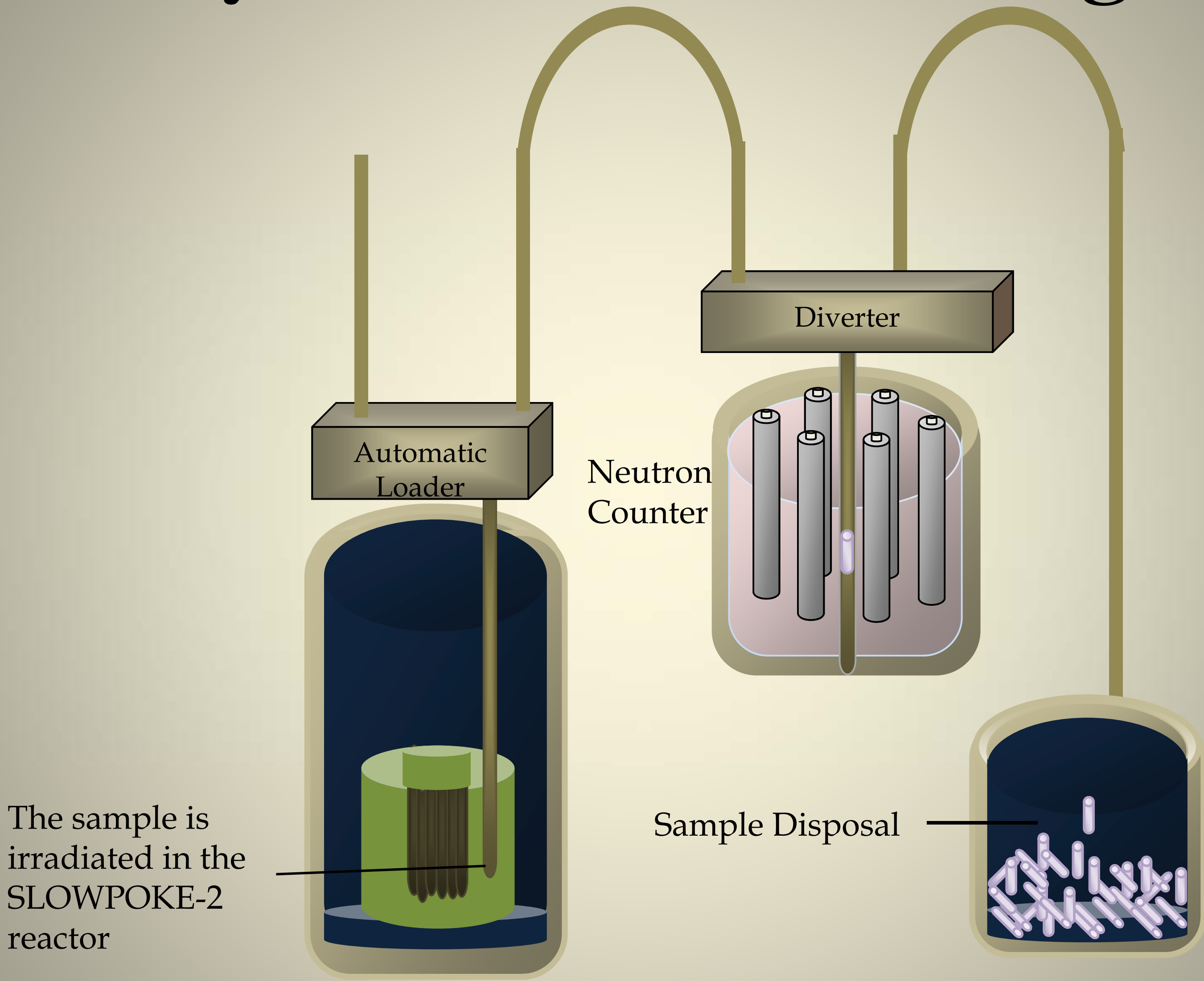


20 kW research reactor enriched to 19.89%

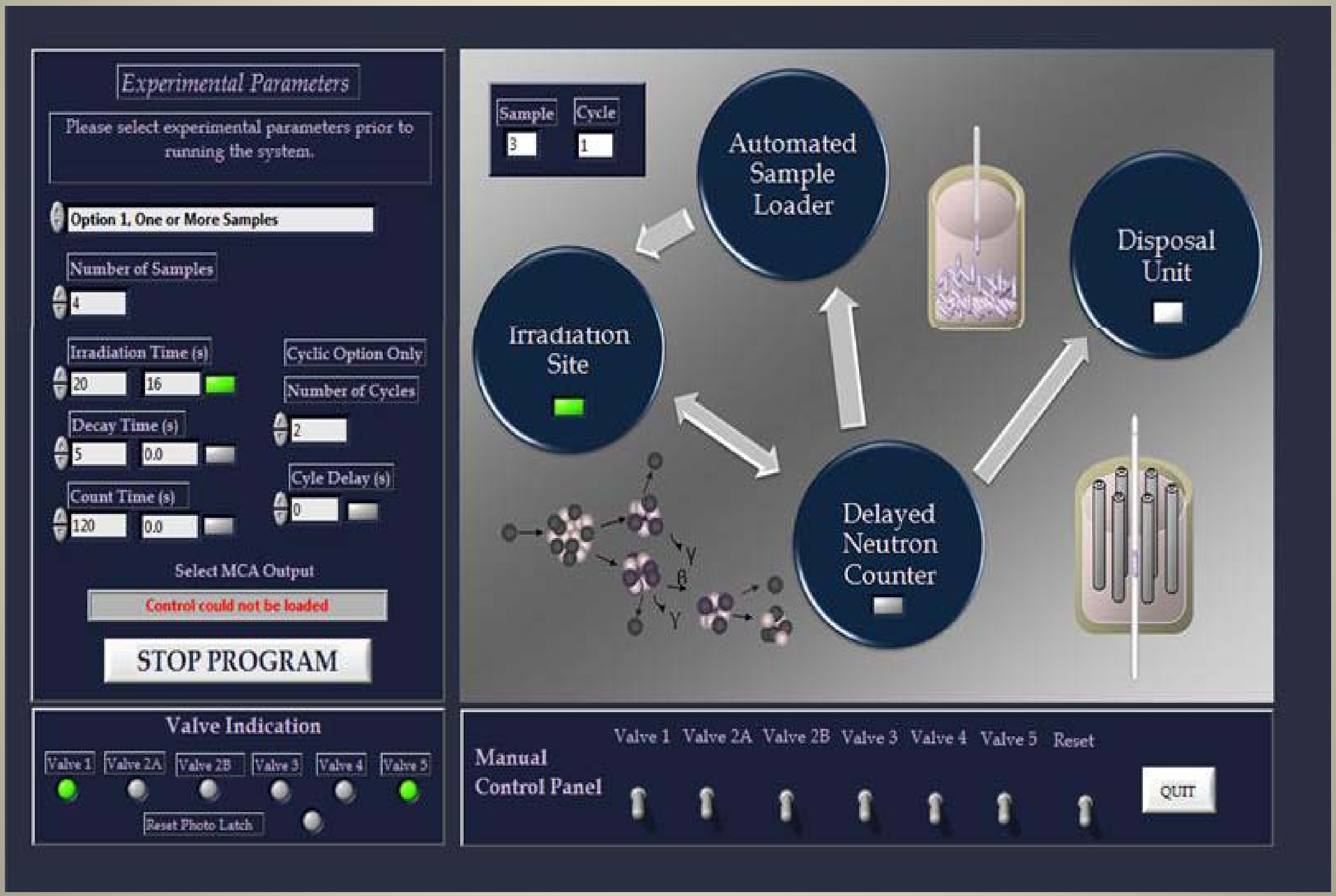
Delayed Neutron Counting Arrangement



The Delayed Neutron Counting System



Hardware & Software Control



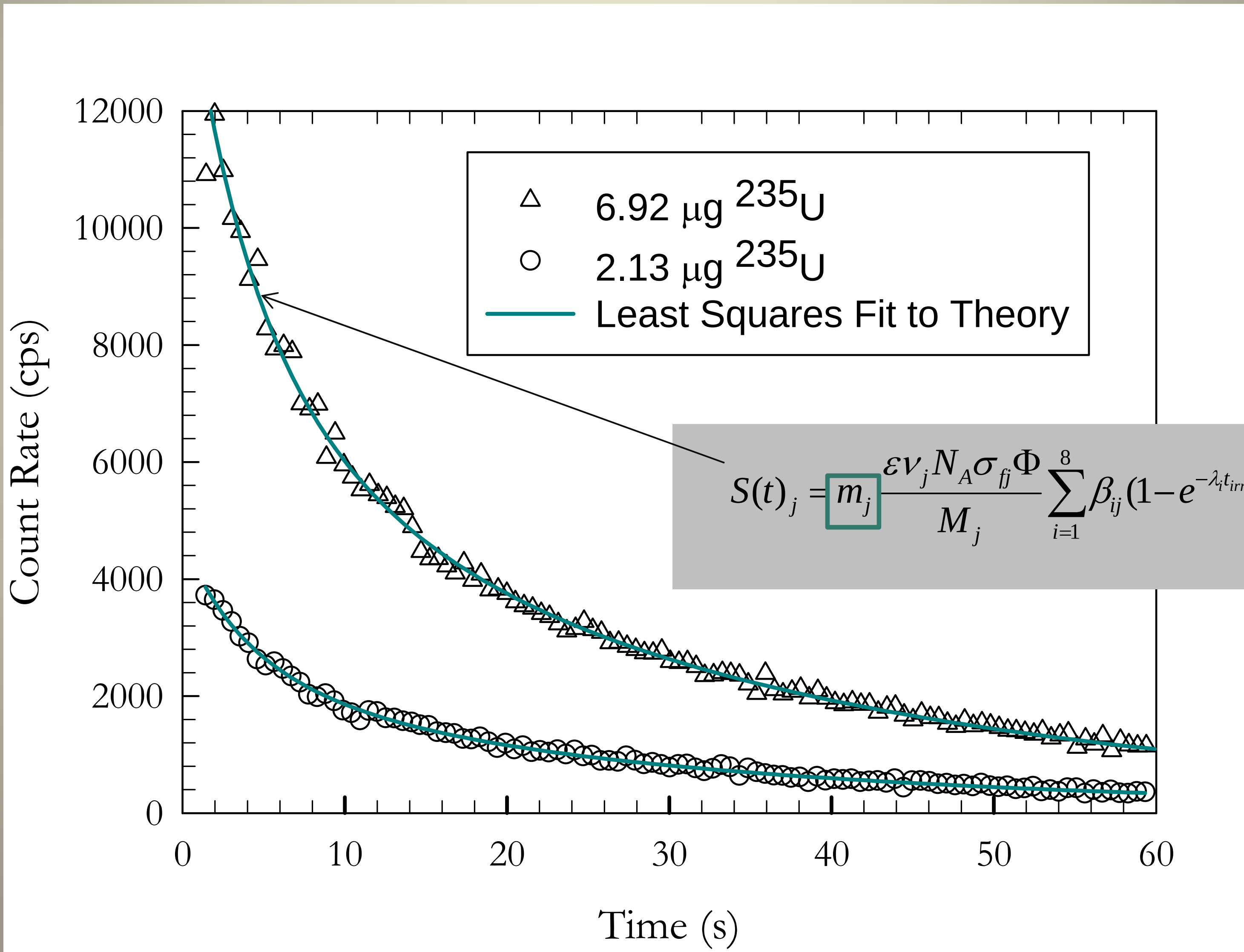
The Fissile Analysis Program

- Imports count excel file:

SLOWPOKE Test Data				
11/04/2011	11:47:49 AM			
Standard? YES				
Sample #	Cycle	Time	Counts	Total Counts
1	A	1.015625	1863	79306
		1.546875	1821	
		2.0625	1759	
		2.59375	1691	

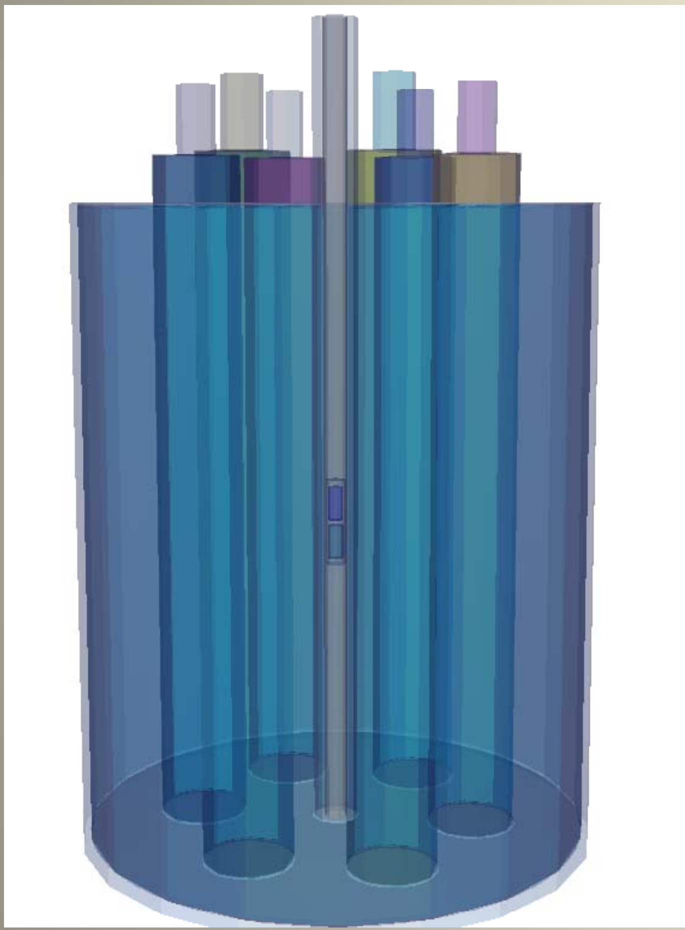
- Corrects for background, dead time, normalizes to counts per second
- Outputs fissile (U-235) content in µg

Fissile Analysis Program Output

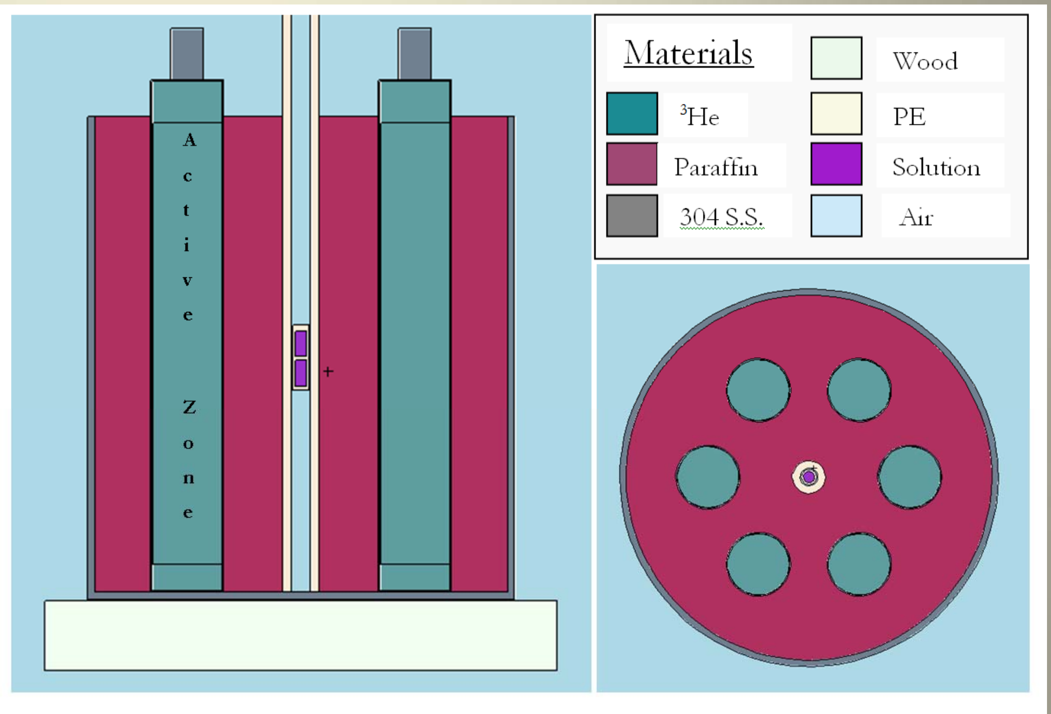


$$S(t)_j = \boxed{m_j}\frac{\varepsilon \nu_j N_A \sigma_{fj} \Phi}{M_j}\sum_{i=1}^{8}\beta_{ij}(1-e^{-\lambda_i t_{irr}})(e^{-\lambda_i t_d})(e^{-\lambda_i t})$$

MCNP6 Geometry

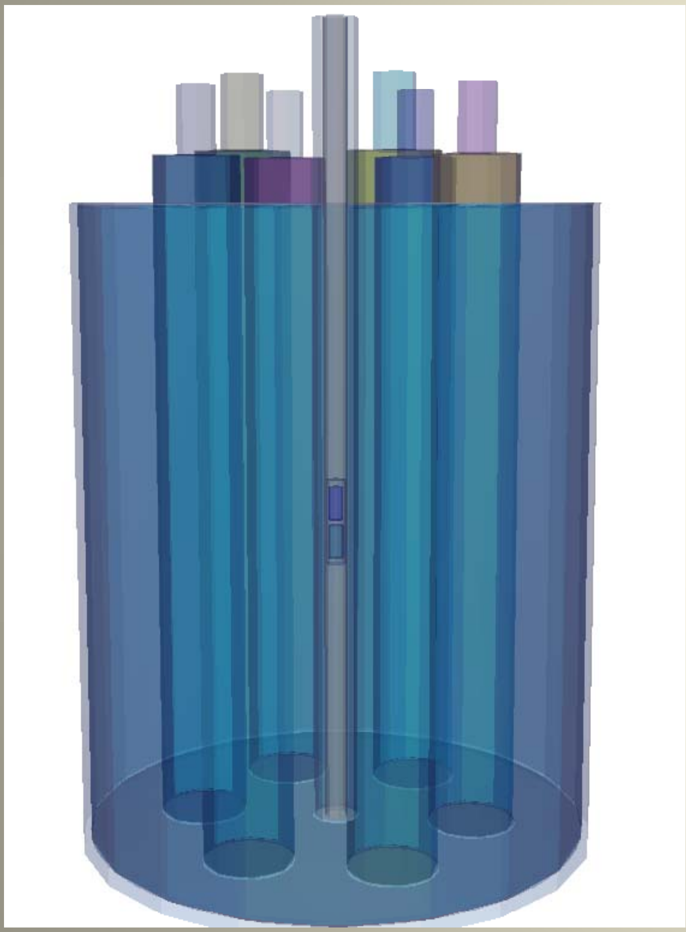


Visual Editor



MCNP6 Plotter

MCNP6 Input Deck Summary



Visual Editor

- tme = 0 -60e8 sh
  - Sample is exposed to defined neutron flux
  - DN activity builds
- tme = 60e8 180e8 sh
  - F8 tallies record delayed neutron activity

MS8 Describe the INPUT DECK
Madison, 10/11/2011

^{3}He Detectors & System Efficiency

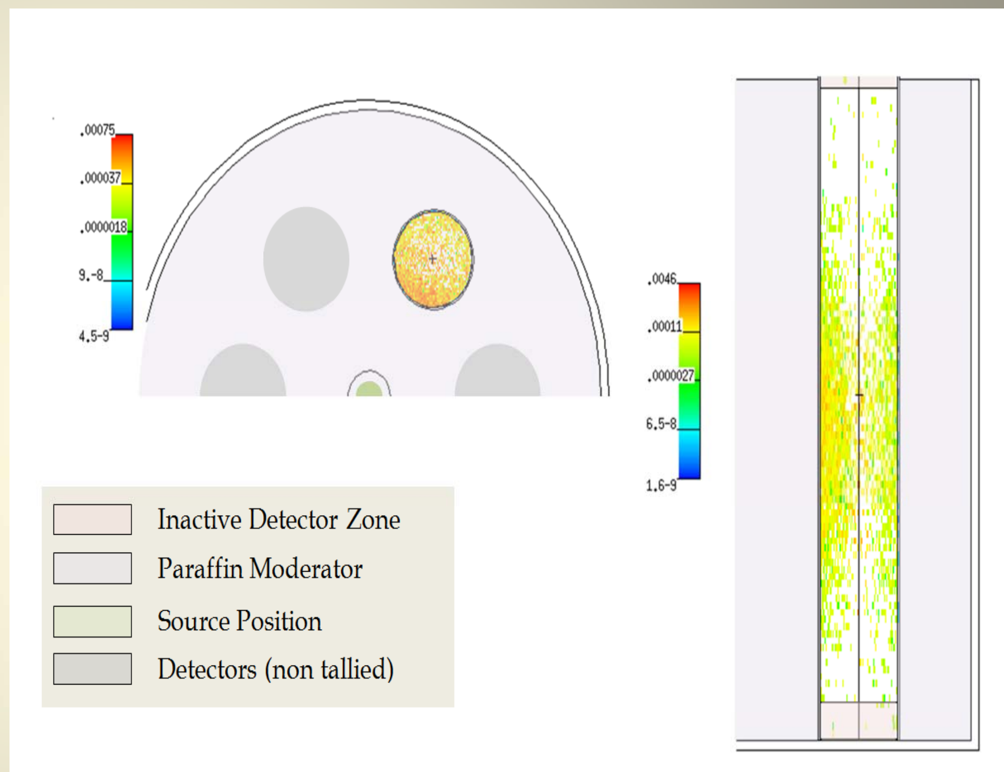
- F8 tallies (pulses) in active zone of detector

- User-defined delayed neutron energies and temporal behaviour using 8-group model

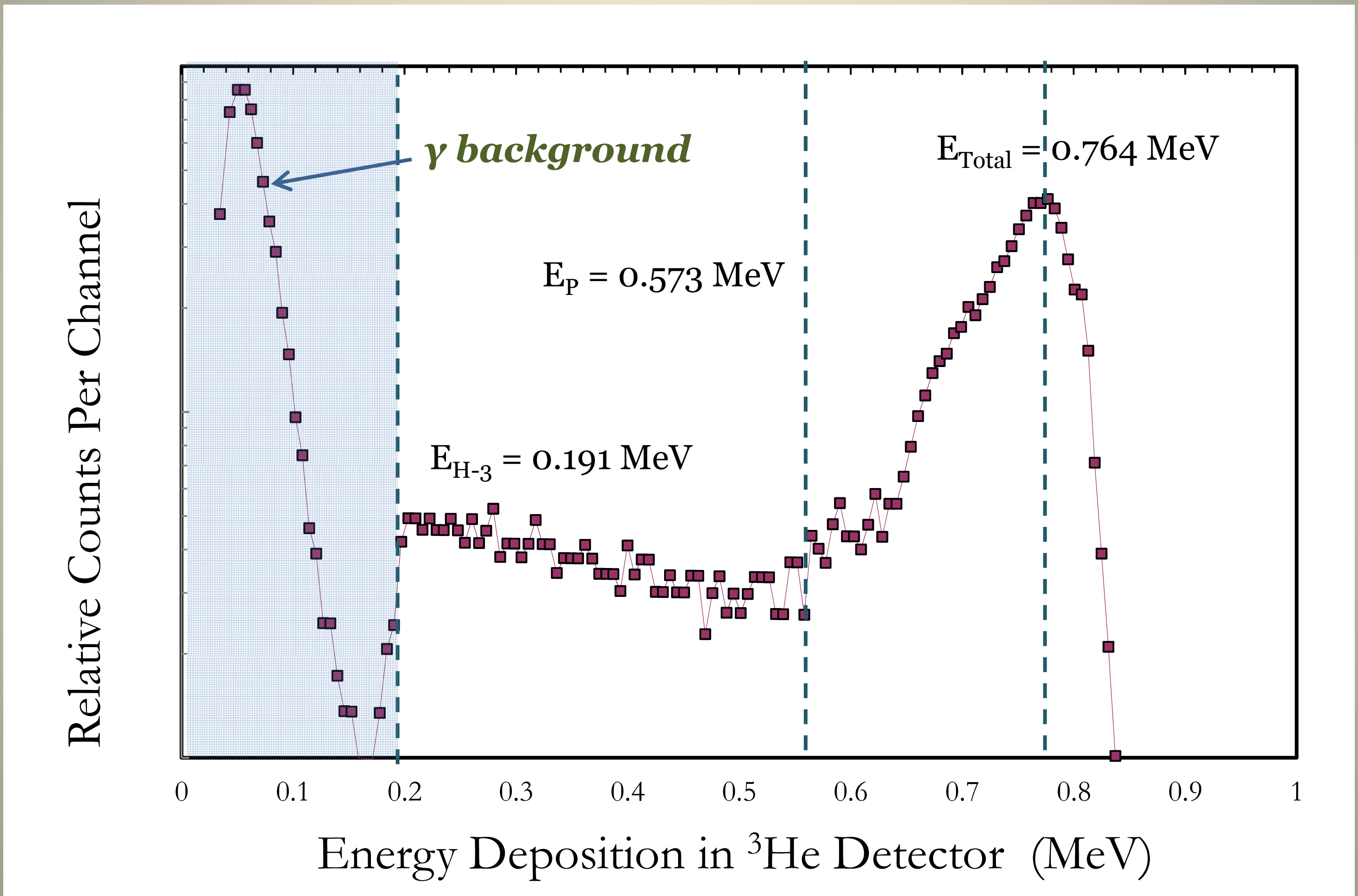
$$S(t)_j = m_j \frac{\boxed{\varepsilon \nu}_j N_A \sigma_{fj} \Phi}{M_j} \sum_{i=1}^{8} \beta_{ij} (1 - e^{-\lambda_i t_{irr}})(e^{-\lambda_i t_d})(e^{-\lambda_i t})$$

- Experimental Efficiency: 34 ± 5%

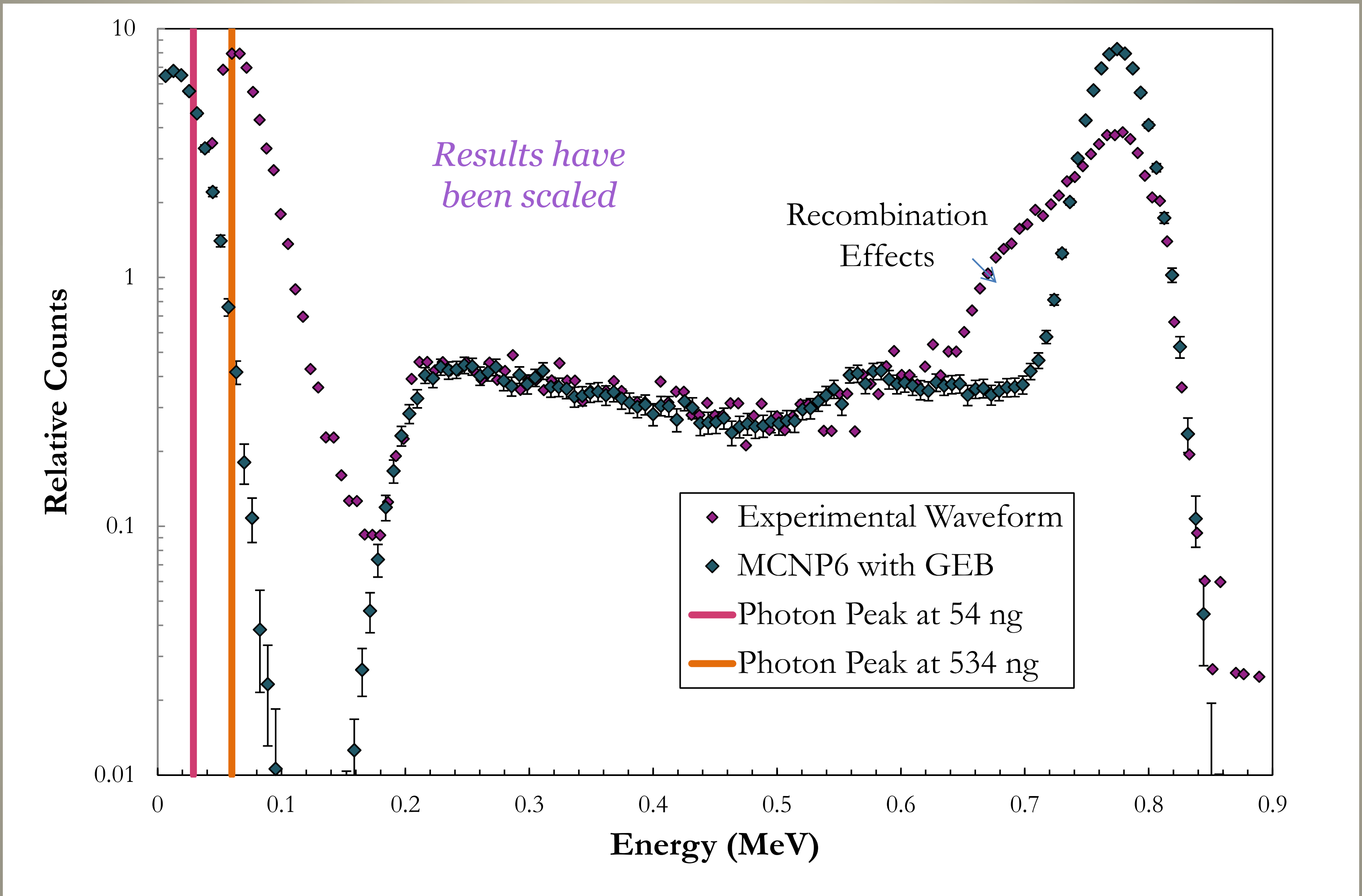
- MCNP6 Efficiency: 37%



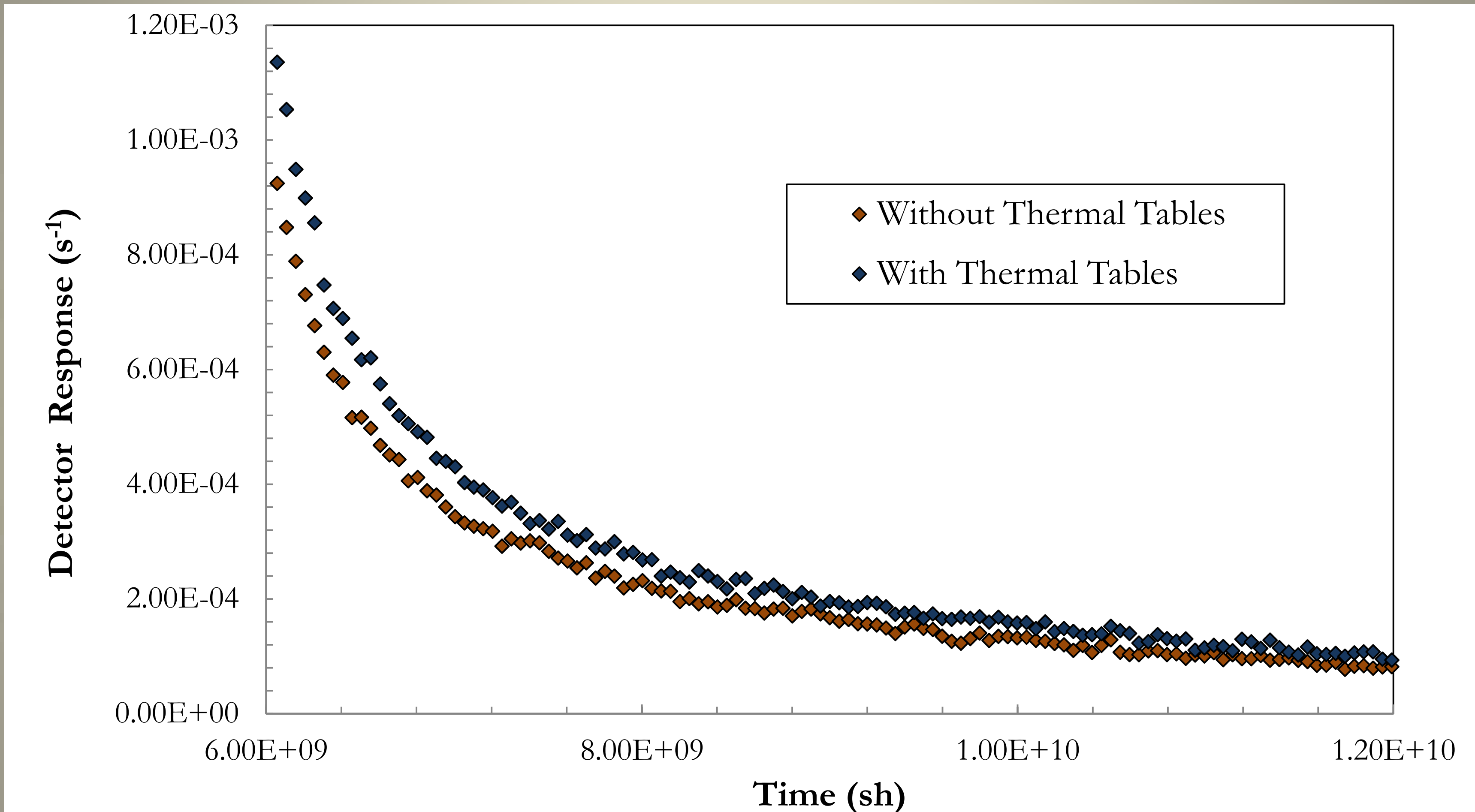
Detector Wall Effects – Experimental Measurements



Detector Wall Effects



S(α,β) for Paraffin & Aqueous Solution



Molecular binding and crystalline effects are important at low neutron energies

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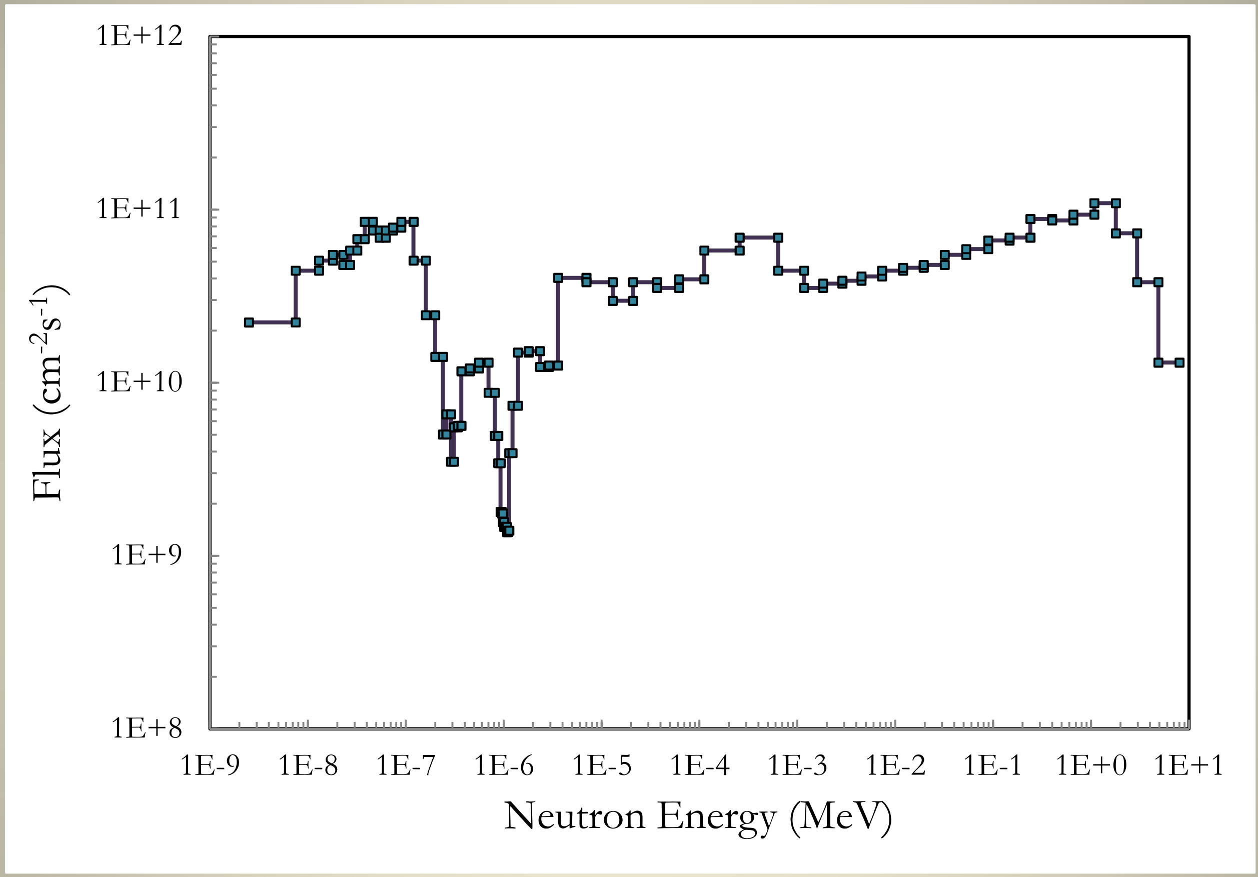
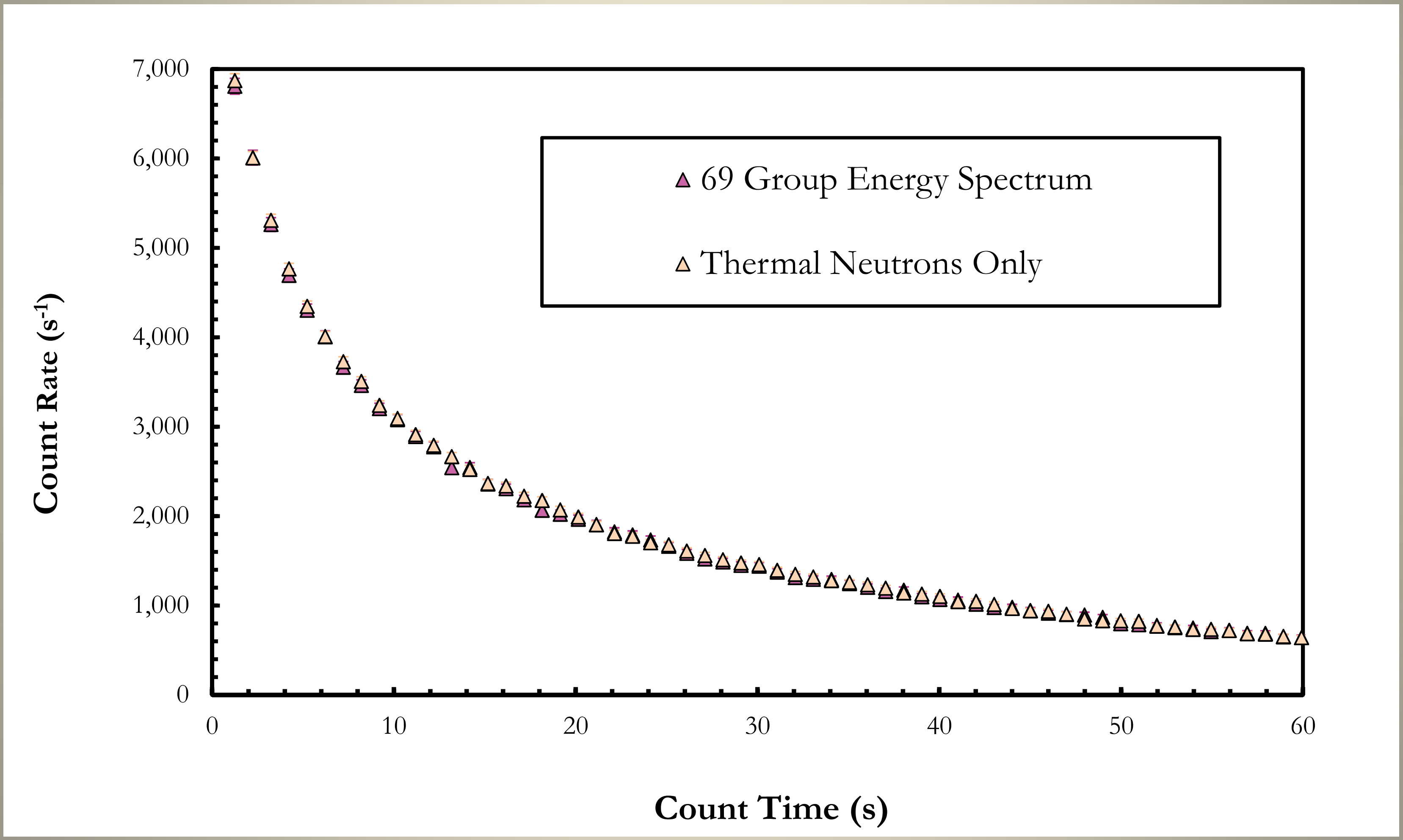
Neutron Flux Estimate in SLOWPOKE-2

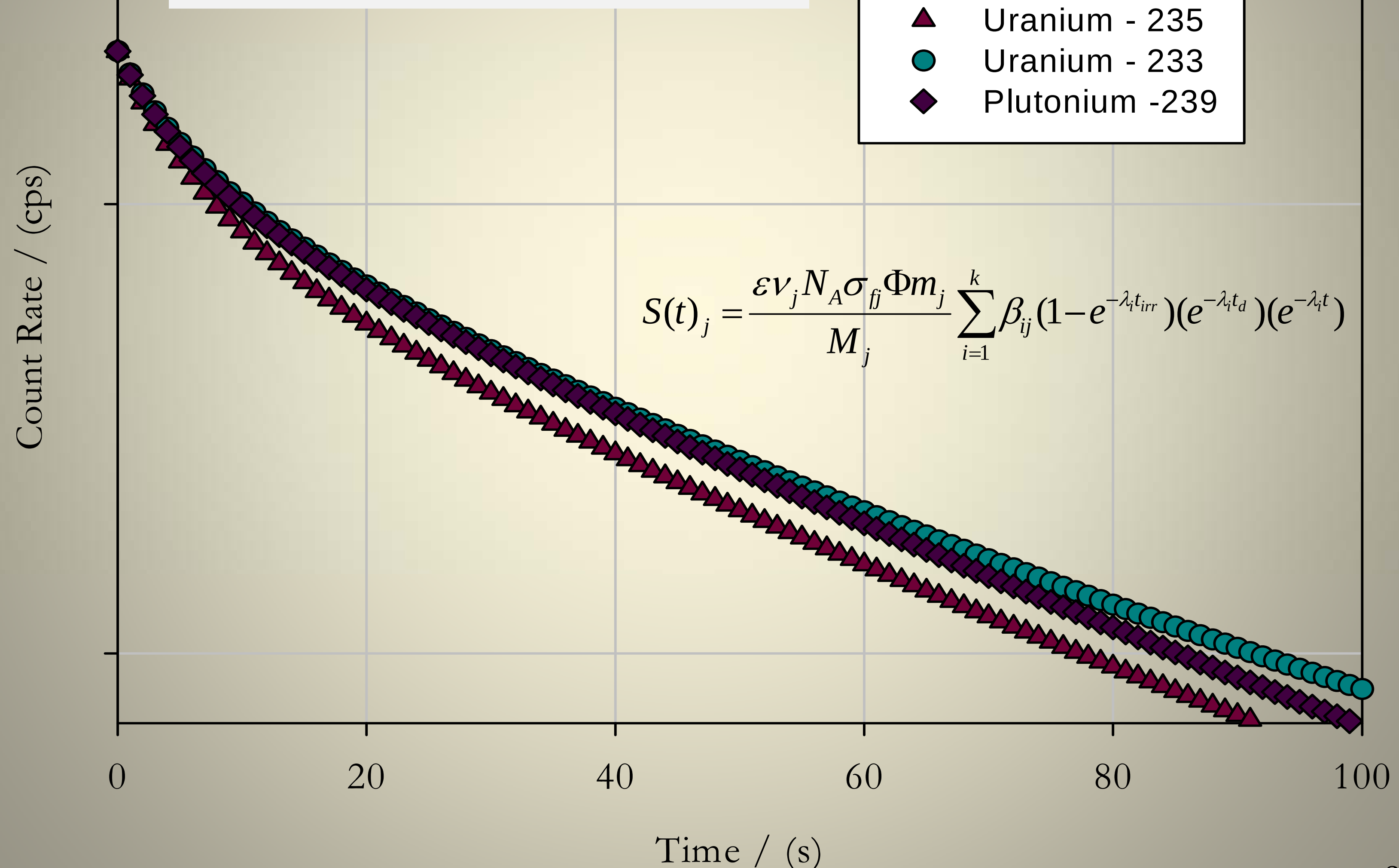


- 69 Group Energy Spectrum from K. Khattab, I. Sulieman, (2010, Syrian MNSR)
- Magnitude of flux determined experimentally in DNC irradiation site

Neutron Flux & DN Behavior

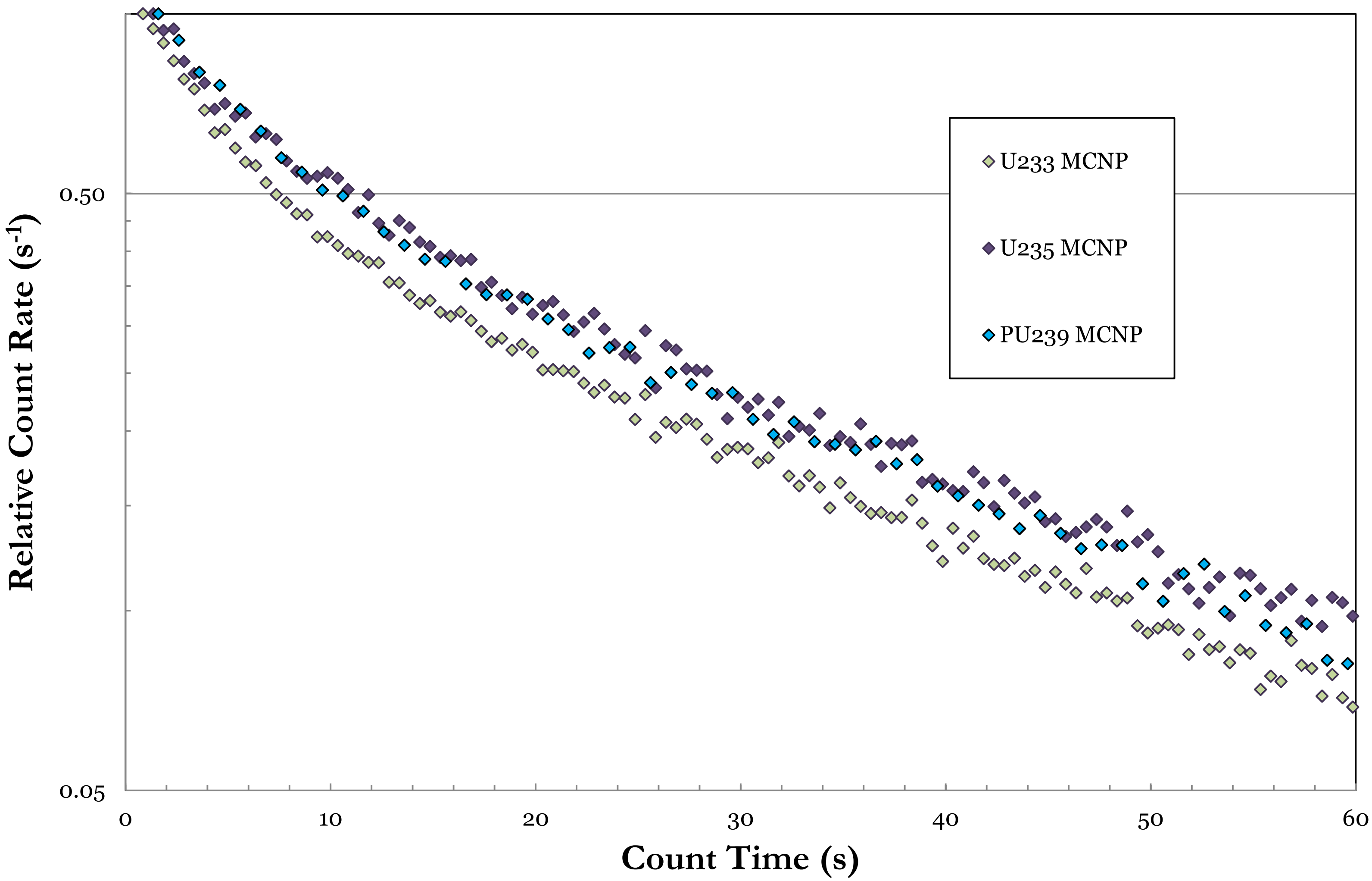


Each fissile isotope has a signature delayed neutron count rate curve

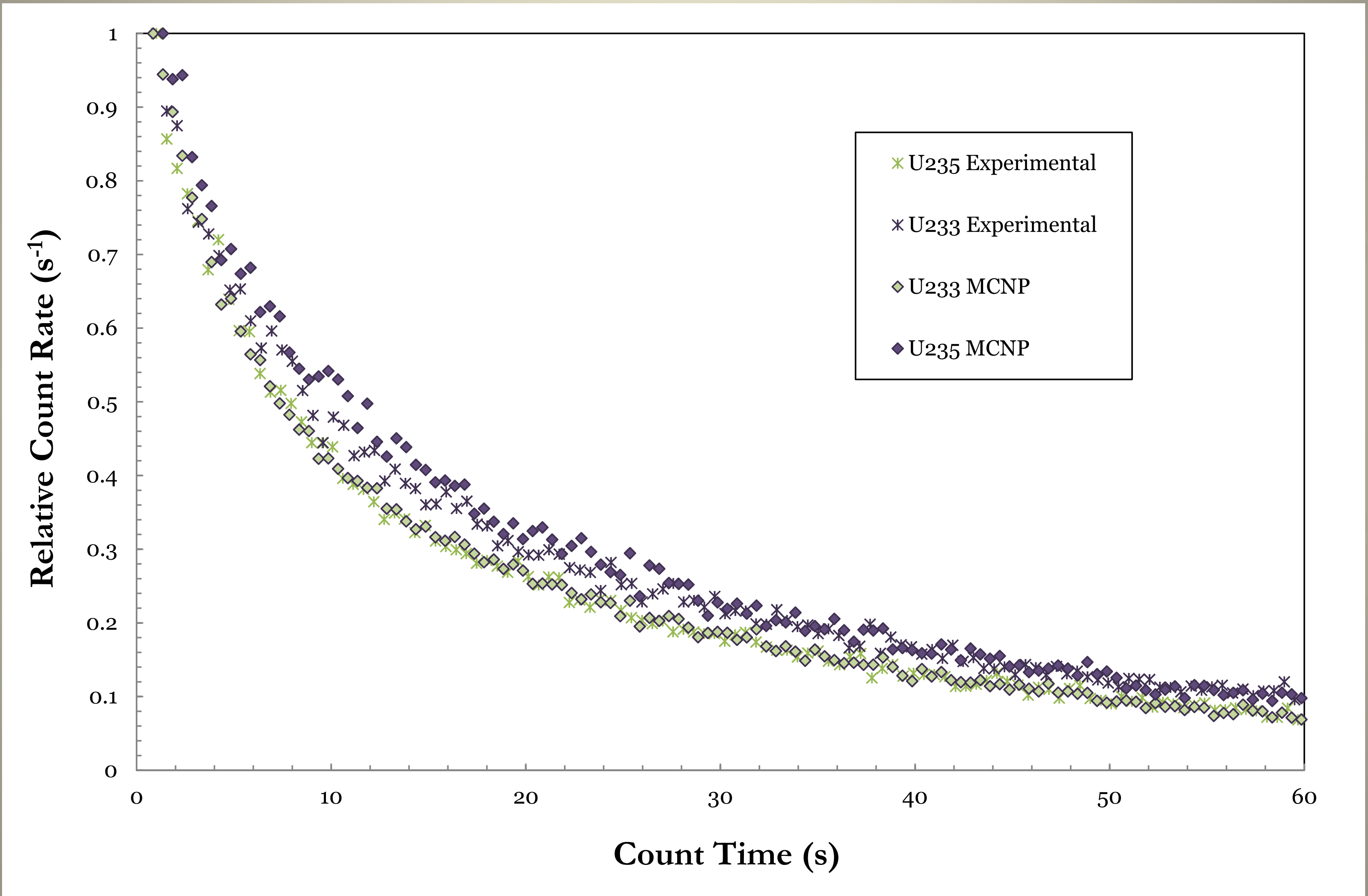


$$S(t)_j = \frac{\varepsilon \nu_j N_A \sigma_{fj} \Phi m_j}{M_j} \sum_{i=1}^{k} \beta_{ij} (1 - e^{-\lambda_i t_{irr}})(e^{-\lambda_i t_d})(e^{-\lambda_i t})$$

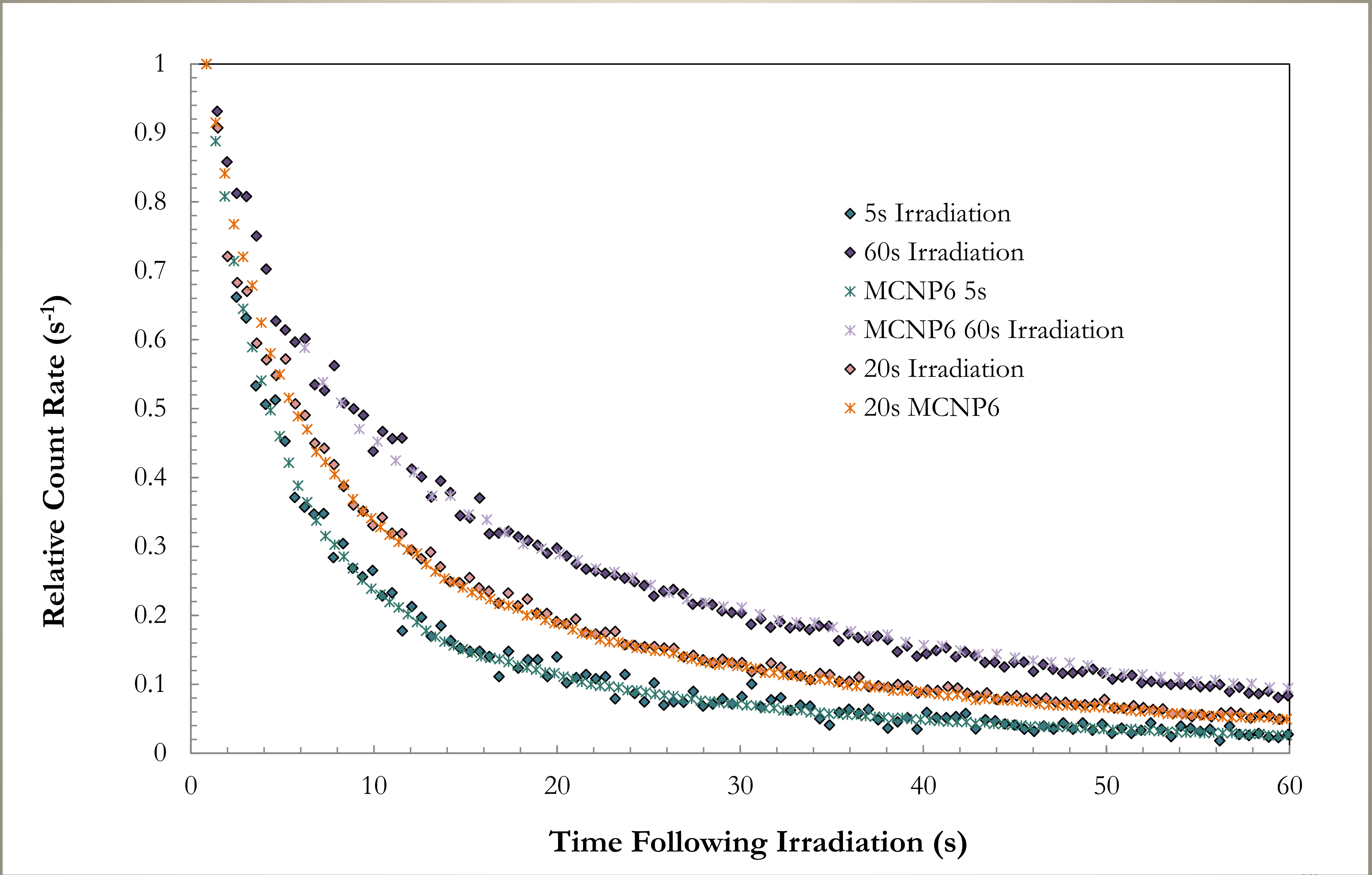
Temporal DN Behavior



Experimental Measurements & MCNP6



Irradiation Duration



CINDER vs. ACE Model

MCNP6 Initial Release Notes:

PHYS:N EMAX EMCNF IUNR DNB unused* FISNU COILF CUTN

DNB =delayed neutron control
-1001 Analog delayed neutrons from ACE tables (if available, otherwise from CINDER tables)
-101 Analog delayed neutrons from CINDER only
–1 Analog delayed neutrons from ACE tables only
0 No delayed neutrons produced
1-15 biased number of neutrons produced per fission.
101-115 Biased number of delayed neutrons from CINDER tables
1001-1015 Biased number of delayed neutrons from ACE tables, otherwise from CINDER tables

CINDER vs. ACE Model

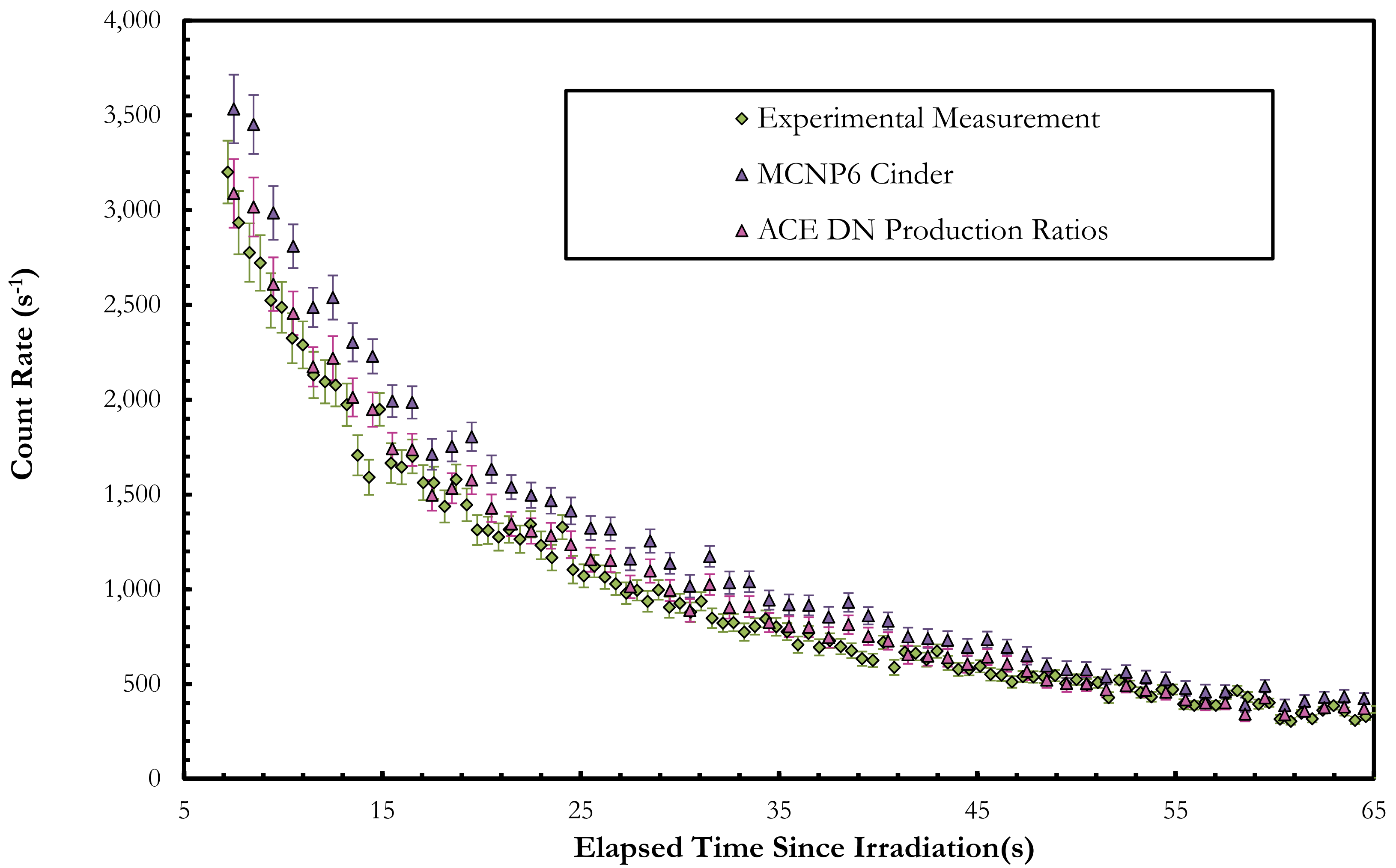
MCNP6 Initial Release Notes:

PHYS:N EMAX EMCNF IUNR DNB unused* FISNU COILF CUTN

DNB: -1001 → ACE; -101 → CINDER

	ACE	CINDER	JEFF [*]
Prompt n	11992	11990	
Delayed n	60	96	
Weight Prompt n	1.52e-5	1.52e-5	1.58e-5
Weight Delayed n	7.61e-8	1.22e-7	1.06e-7

Un-normalized Comparisons – ^{235}U ($\pm 1\sigma$)

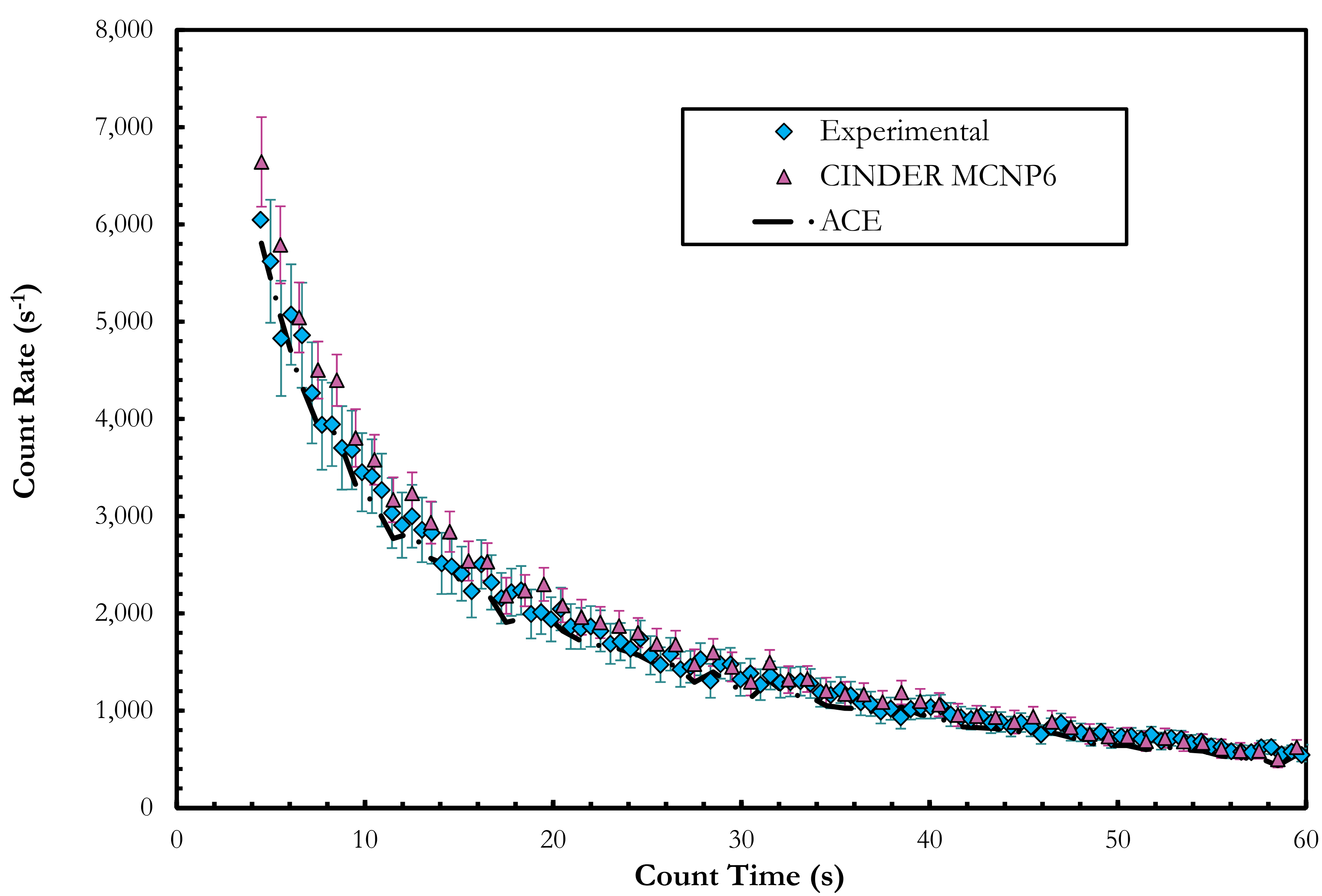


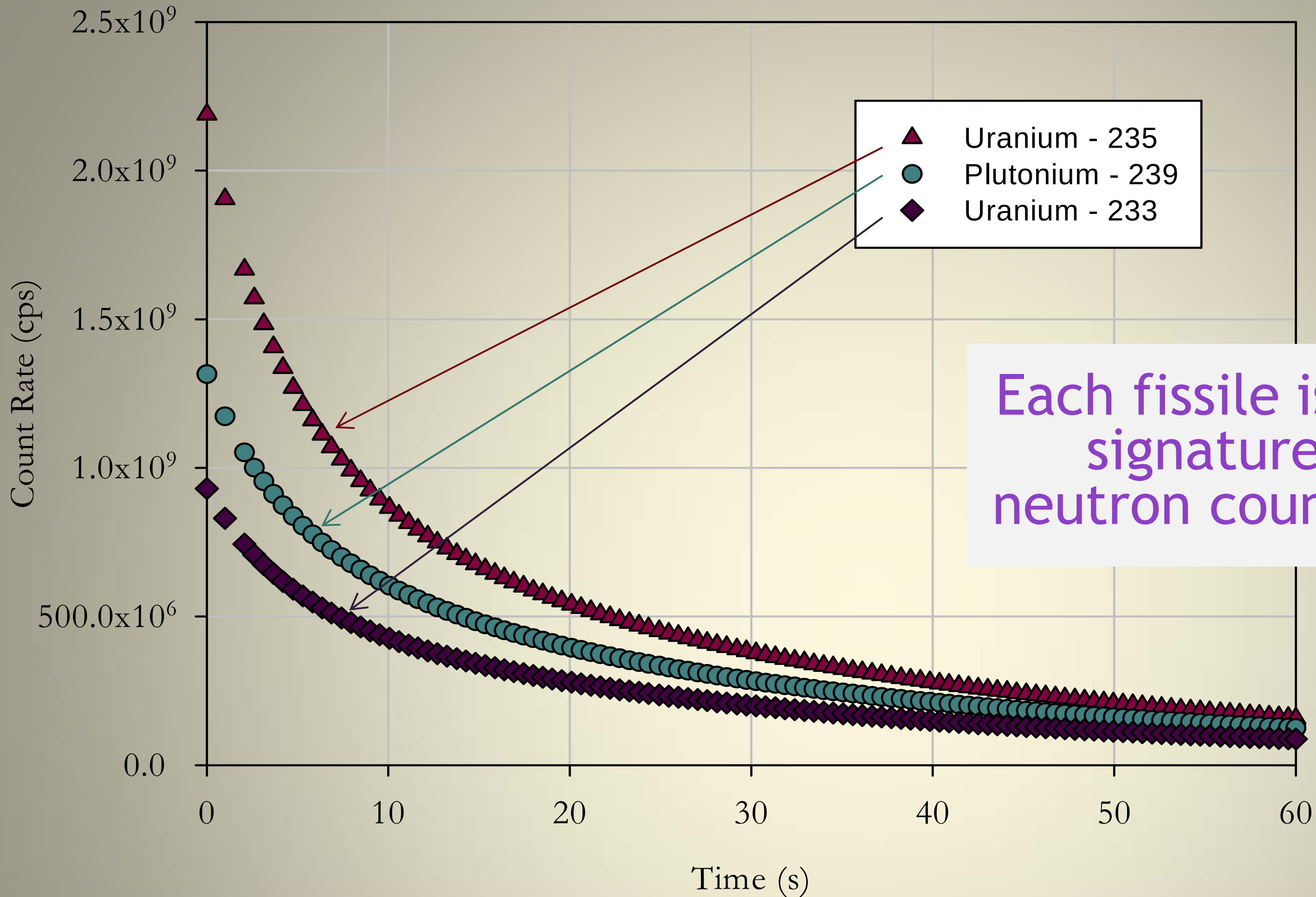
Future Work

- Continued DNC experimentation and MCNP6 modeling
  - Longer Count Times
  - Plutonium-239 & Uranium-233
  - Mixtures of multiple fissile isotopes
- Delayed Gamma Emission from Special Nuclear Materials
  - System to be built at RMC late 2011/2012
  - To be modeled in MCNP
- Work to be presented and submitted to ANS Conference in 2012

Thank you

Un-normalized Comparisons – ^{235}U





Each fissile isotope has a signature delayed neutron count rate curve

Isotope j Count Rate:

$$S(t)_j = \frac{\varepsilon \nu_j N_A \sigma_{fj} \Phi m_j}{M_j} \sum_{i=1}^{k} \beta_{ij} (1 - e^{-\lambda_i t_{irr}})(e^{-\lambda_i t_d})(e^{-\lambda_i t})$$

n isotopes:

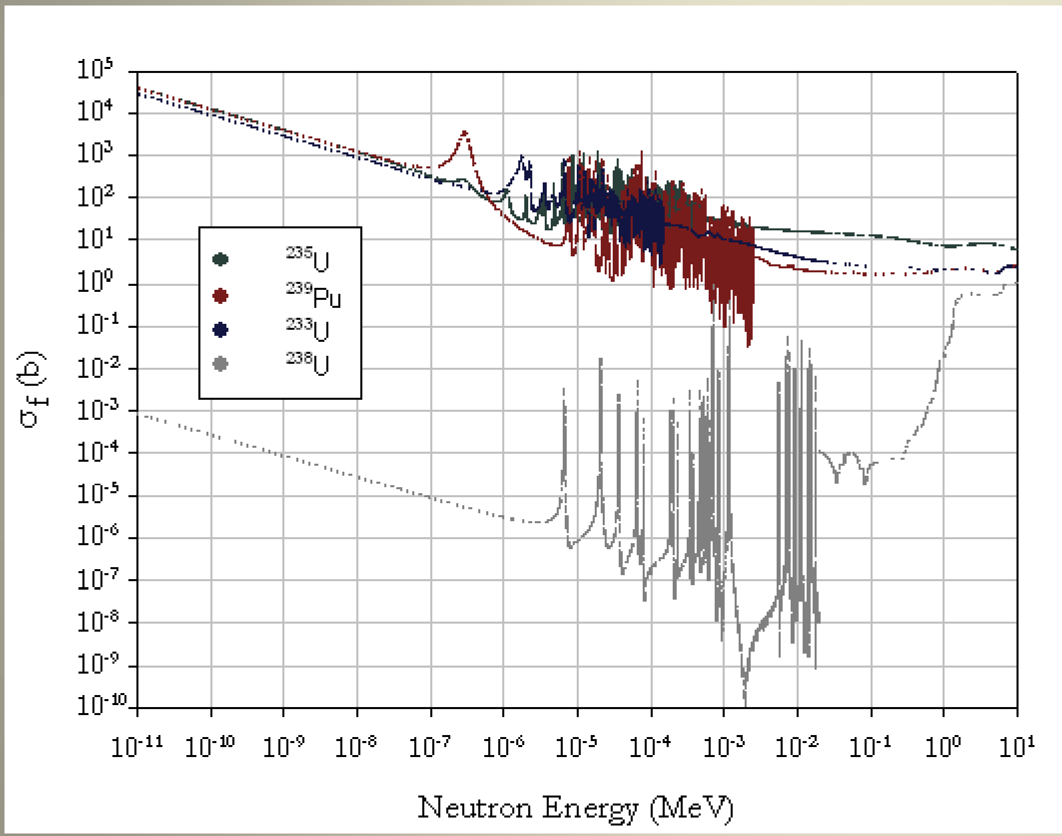
$$S(t) = \sum_{j=1}^{n} S(t)_j$$

Total Count Rate:

$$C(t) = S(t) + Background$$

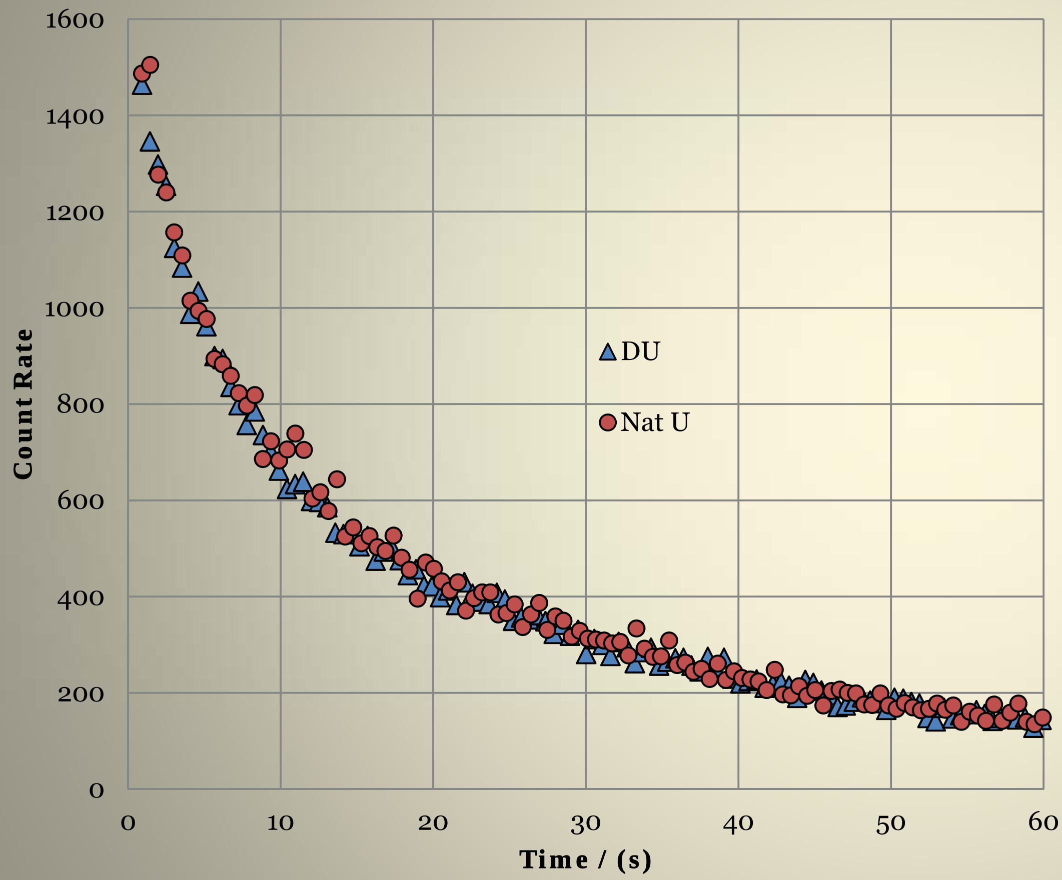
ε = system efficiency
ϕ = neutron flux
λ_i = lifetime of group i
υ_j = delayed n production
m_j = mass of isotope
t_{irr} = irradiation time
N_A = Avogadro's number
M_j = Molar Mass of isotope
t_d = decay time
σ_f = fission cross section
β_i =production ratio for group i
t = count time

^{238}U Fission Contributions



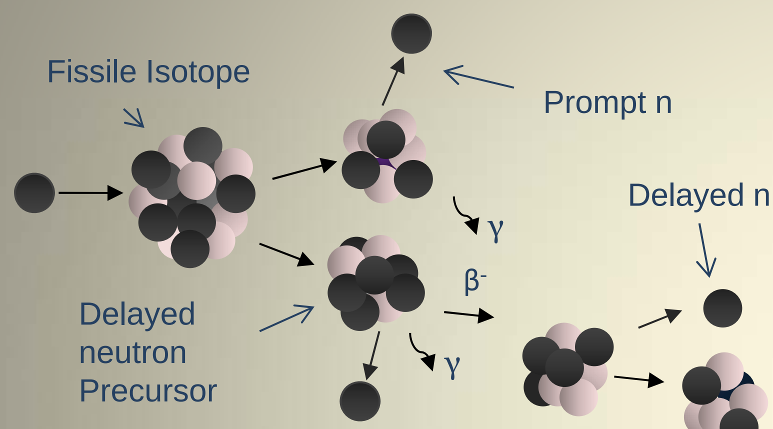
- Used epi/thermal ratio from Andrews thesis
- Account for flux, delayed neutron production, isotopic composition of U samples
- Determined that for natural U < 0.001% of DN recorded are from the fission of ^{238}U

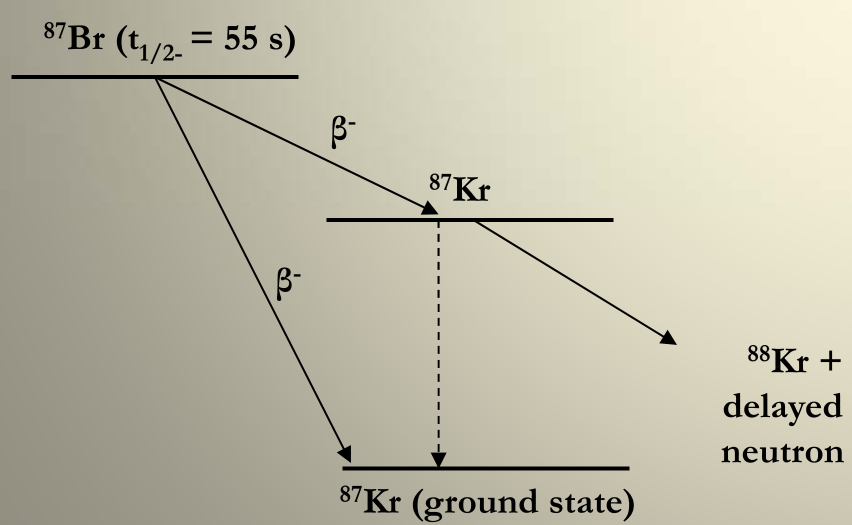
^{238}U Fission Contributions

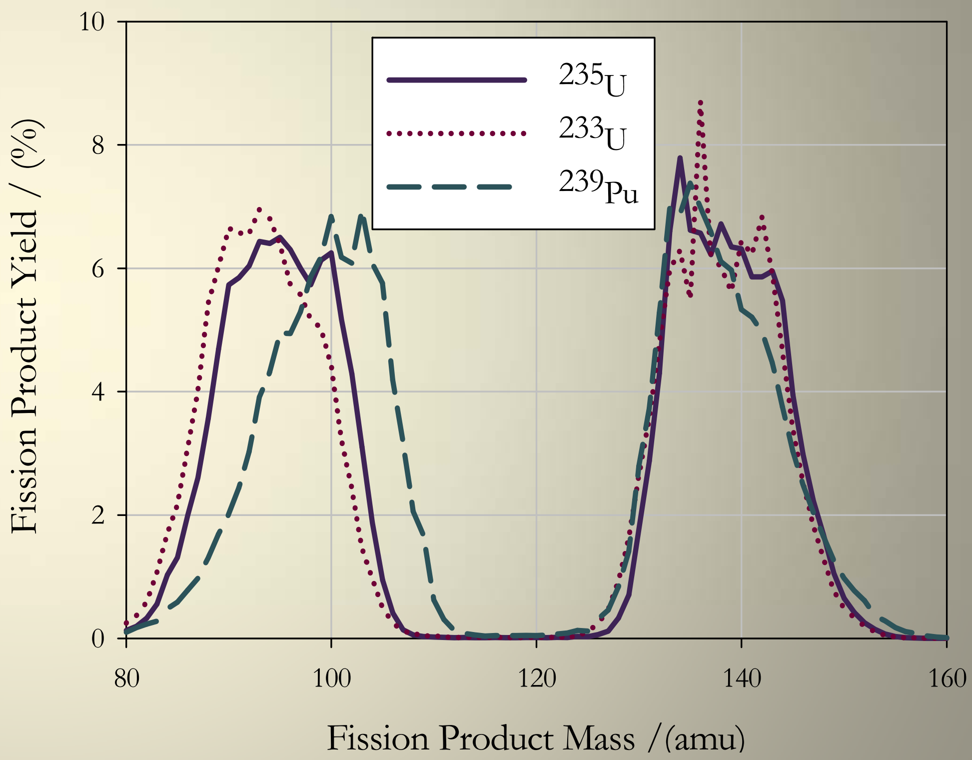


- Samples contain identical amounts of ^{235}U
- DU sample has ~1.35x the ratio of ^{238}U to ^{235}U
- Appears to have no effect on the magnitude of the count rate curve

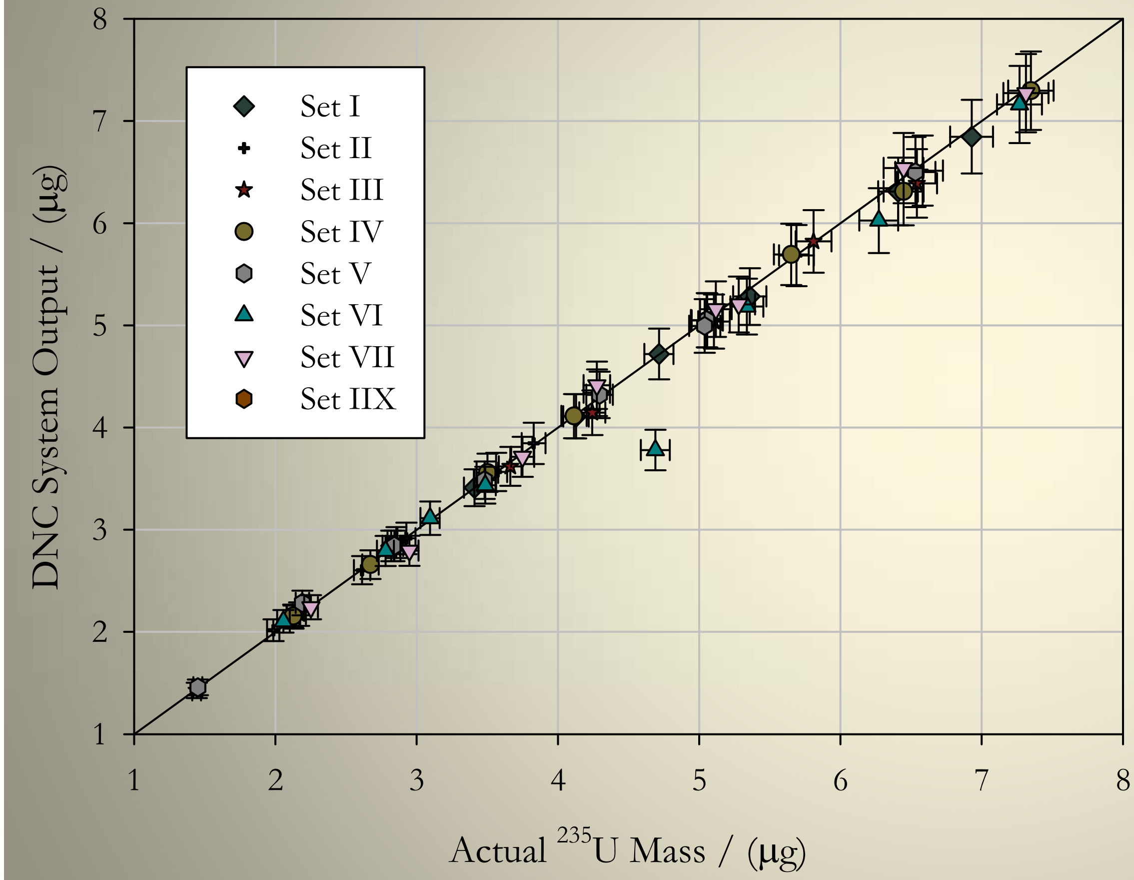
Fission Fragment Yield & Delayed Neutron Precursors







System Reproducibility – Natural Uranium



Ideal: Slope = 1.0 Y-Intercept = 0.0

Set	Slope (2σ uncertainty)	Y-Intercept (µg) (2σ uncertainty)
I	0.98 ± 0.01	0.08 ± 0.04
II	0.96 ± 0.08	0.1 ± 0.4
III	1.00 ± 0.02	-0.02 ± 0.06
IV	1.0 ± 0.1	0.04 ± 0.06
V	0.98 ± 0.01	0.08 ± 0.04
VI	0.96 ± 0.08	0.1 ± 0.3
VII	1.00 ± 0.02	0.0 ± 0.1
IIX	0.99 ± 0.01	0.04 ± 0.06

System Validation – Depleted Uranium

Sample	Actual ^{235}U Mass (μg)	DNC System ^{235}U Determination (μg)	Relative Error
1	5.52 ± 0.06	5.3 ± 0.2	-3.45%
2	5.56 ± 0.06	5.3 ± 0.2	-5.35%
3	5.50 ± 0.06	5.3 ± 0.2	-3.01%
4	5.56 ± 0.06	5.3 ± 0.2	-4.32%
5	5.53 ± 0.06	5.3 ± 0.2	-4.78%
6	5.62 ± 0.06	5.4 ± 0.2	-4.45%
7	5.59 ± 0.06	5.4 ± 0.2	-4.24%
8	5.53 ± 0.06	5.3 ± 0.2	-4.47%

Relative Error
-3.6%

Actual ^{235}U mass (μg)	Total Solution Mass (g)	Experimental Mass (g)	Relative Error (%)
1.54 ± 0.04	0.290 ± 0.003	1.5 ± 0.1	-3.2
1.54 ± 0.04	0.379 ± 0.003	1.5 ± 0.1	-3.2
1.54 ± 0.04	0.568 ± 0.003	1.5 ± 0.1	-2.6
1.55 ± 0.04	0.660± 0.003	1.6 ± 0.1	-0.6
1.53 ± 0.04	0.819 ± 0.003	1.5 ± 0.1	-3.9
1.46 ± 0.04	0.738 ± 0.003	1.5 ± 0.1	-1.4
1.59 ± 0.04	0.943 ± 0.003	1.5 ± 0.1	-3.8

Error independent of solution nitric acid solution volume

Determining ^{235}U Content